

Light Water Reactor Sustainability Program

Materials Aging and Degradation Technical Program Plan



August 2016

U.S. Department of Energy

Office of Nuclear Energy

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Light Water Reactor Sustainability Program
Materials Aging and Degradation Pathway
Technical Program Plan

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August 2016

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managed by
UT-BATTELLE, LLC
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

EXECUTIVE SUMMARY

Components serving in a nuclear reactor plant must withstand a very harsh environment including extended time at temperature, neutron irradiation, stress, and/or corrosive media. The many modes of degradation are complex with synergies between multiple environmental variables and conditions that vary depending on location and material. However, understanding and managing materials degradation is a key for the continued safe and reliable operation of nuclear power plants.

Extending reactor service to beyond 60 years will increase the demands on materials and components. Therefore, an early evaluation of the possible effects of extended lifetime is critical. The recent report NUREG/CR-7153 (volumes 1 -5) gives a detailed assessment of many of the key issues in today's reactor fleet and provides a starting point for evaluating those degradation forms particularly important for consideration in extended lifetimes. While life beyond 60 will add additional time and neutron fluence, the primary impact will be increased susceptibility (although new mechanisms are possible).

For reactor pressure vessels (RPVs), a number of significant issues have been recommended as deserving attention in future research activities. Large predictive uncertainties for embrittlement can result from limited data available at high fluences, for long times, and for differences in alloy solute concentrations. The use of test reactors at high fluxes to obtain high fluence data is problematic for representation of the low flux conditions in RPVs. The so-called "late-blooming phases" of Mn-Ni-Si enriched particles, especially for high-nickel welds, have been observed, and additional experimental data are needed in the high fluence regime where they are expected.

For the reactor core and primary systems, several key areas have been identified. Thermo-mechanical considerations such as aging and fatigue must be examined. Irradiation-induced processes must also be considered for higher fluences, particularly the influence of radiation-induced segregation, swelling, and/or precipitation on overall materials performance. Corrosion takes many forms within the reactor core, although irradiation-assisted stress corrosion cracking is of the highest interest in extended life scenarios. Environmentally assisted fatigue is another area for which more research is needed. Research in these areas can build upon other ongoing programs in the light water reactor (LWR) industry as well as other reactor materials programs (such as fusion and fast reactors) to help resolve these issues for extended LWR life.

In the secondary systems, corrosion is extremely complex. Understanding the various modes of corrosion and identifying mitigation strategies are important steps for long-term service. Primary water stress corrosion cracking is a key form of degradation in extended service scenarios.

In the area of welding technology, two critical long-standing welding-related technical challenges requiring further research and development (R&D), both fundamental and applied. The first is the need for an advanced weld simulation tool to support component

life extension and reliable lifetime prediction, especially as related to the issue of residual stresses as a primary driving force for stress corrosion cracking. The second challenge is the development of new welding technologies for reactor repair and upgrade.

Concrete structures can also suffer undesirable changes with time because of improper specifications, a violation of specifications, or adverse performance of its cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete's service life. Research is needed in a number of areas to ensure the long-term integrity of the reactor concrete structures. For example, radiation effects on containment concrete emerged as the most important degradation mechanism, mainly driven by insufficient data to improve the level of knowledge about the effects of irradiation on concrete mechanical properties. Alkali-aggregate reaction (AAR), acid attack and creep emerged as secondarily important mechanisms. The biggest surprise in this analysis is the result that susceptibility to fracture emerged as the least important mechanism. This should be interpreted to apply only to concrete cracking of the generally known type that is accounted for in the structural design.

Reliability and assurance of the performance of instrumentation and control cable is another important area of concern. Environmental stressors that include radiation, moisture, temperature, oxygen content, mechanical stresses that include tension, compression and vibrational effects, influence long-term performance of cables. Research is required in areas of understanding long term synergistic effects of the environmental variables, inverse temperature effects, accurate methods of determining activation energies for degradation modes, and the effects of dose rate and diffusion limited oxidation. New methods for cable condition monitoring are also required.

Clearly, materials degradation will impact reactor reliability, availability, and, potentially, safe operation. Routine surveillance and component replacement can mitigate these factors; however, failures still occur. With reactor life extensions up to 60 years or beyond and power uprates, many components must tolerate more demanding reactor environments for even longer times. This may increase susceptibility to degradation for different components and may introduce new degradation modes. While all components (except perhaps the RPVs) can be replaced, it may not be economically favorable. Therefore, understanding, controlling, and mitigating materials degradation processes and a technical basis for long-range planning for necessary replacements are key priorities for reactor operation, power uprate considerations, and life extensions.

Many of the various degradation modes are highly dependent on a number of different variables, creating a complex scenario for evaluating lifetime extensions. To resolve these issues for life extension, a science-based approach is critical. Modern materials science tools (e.g., advanced characterization tools, past experience, and computational tools) must be employed. To effectively address gaps in the scientific understanding of materials behavior, different methodologies that include experimental testing, computation modeling and analysis of harvested materials should be included in the research activities.

This is in part due to the differences that may arise between long term/low flux aging versus accelerated high flux testing. Ultimately, safe and efficient extension of reactor service life will depend on progress in several distinct areas, including mechanisms of degradation, mitigation strategies, modeling and simulations, monitoring, and management

The Materials Aging and Degradation (MAaD) task within the Light Water Reactor Sustainability (LWRS) program is charged with R&D to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear reactors. The work will provide data and methods to assess performance of systems, structures, and components (SSCs) essential to safe and sustained reactor operations. The research and development outcomes produced from this program will be used to inform utilities, industry groups, and regulators on operational and regulatory requirements for materials in reactor SSCs subjected to long-term operation conditions.

The objectives of this report are to describe the motivation and organization of the MAaD pathway within the LWRS program, provide detail on the individual research tasks within MAaD, describe the outcomes and deliverables of MAaD, including recent technical highlights and progress, and list the requirements for performing this research.

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ACRONYMS

ANL	Argonne National Laboratory
BAC	boric acid corrosion
BWR	boiling water reactor
CASS	cast austenitic stainless steel
CVN	Charpy V notch
dpa	displacements per atom
DOE	US Department of Energy
EAC	environmentally assisted cracking
EMDA	Expanded Materials Degradation Assessment
EPRI	Electric Power Research Institute
FAC	flow-accelerated corrosion
FY	fiscal year
I&C	instrumentation and control
IASCC	irradiation-assisted stress corrosion cracking
IGSCC	intergranular stress corrosion cracking
INL	Idaho National Laboratory
LTCP	low temperature crack propagation
LTO	Long-Term Operations (EPRI program)
LWR	light water reactor
LWRS	Light Water Reactor Sustainability (DOE program)
MAaD	materials aging and degradation
MBI	microbiologically influenced
MDM	materials degradation matrix
NDE	nondestructive examination
NE	Office of Nuclear Energy (of DOE)
NRC	US Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PMDA	proactive materials degradation assessment
PNNL	Pacific Northwest National Laboratory
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
R&D	research and development

RCS	reactor coolant system
RIS	radiation-induced segregation
RPV	reactor pressure vessel
SCC	stress corrosion cracking
SSC	system, structure, and component
TGSCC	transgranular stress corrosion cracking

Light Water Reactor Sustainability Program Integrated Program Plan

1. BACKGROUND

Nuclear power currently provides almost 20% of the electrical power generation and almost 63% of the non-carbon-emitting power generation in the United States. In future years, nuclear power must continue to generate a significant portion of the nation's electricity to meet growing electricity demand and clean energy goals, and to ensure energy independence. New reactors will be an essential part of nuclear power expansion, but given the limits on new builds imposed by economics and industrial capacity, existing light water reactors must also be managed for extended service.

Ensuring public safety and environmental protection is a prerequisite to all nuclear power plant operating and licensing decisions at all stages of reactor life. This includes the original license period of 40 years, the first license extension to 60 years, and certainly for any consideration of life beyond 60 years. For extended operating periods, it must be shown that adequate aging management programs are present or planned and that appropriate safety margins exist throughout license renewal periods. Unfortunately, nuclear reactors present a very harsh environment for component service. Materials degradation can impact reactor reliability, availability, plant economic viability, and, potentially, safe operation. Components within a reactor must tolerate the harsh environment of high temperature water, stress, vibration, and, for those components in the reactor core, an intense neutron field. Degradation of materials in this environment can lead to reduced performance over time. Clearly, understanding materials degradation and accounting for the effects of a reactor environment in operating and regulatory limits are essential.

Materials degradation in a nuclear power plant is extremely complex due to the various materials, environmental conditions, and stress states. Over 25 different metal alloys can be found within the primary and secondary systems (Figure 1 [1]); additional materials exist in concrete, the containment vessel, instrumentation and control equipment, cabling, buried piping, and other support facilities. Dominant forms of degradation can vary greatly between different systems, structures, and components (SSCs) in the reactor and can have an important role in the safe and efficient operation of a nuclear power plant. When this diverse set of materials is placed in a complex and harsh environment, coupled with load and degradation over an extended life, an accurate estimate of the changing material behaviors and lifetime is complicated. To address this issue, the US Nuclear Regulatory Commission (NRC) has developed a Progressive Materials Degradation Approach (PMDA) described in NUREG/CR-6923 [2]. The Electric Power Research Institute (EPRI) has utilized a similar approach to develop their own Materials Degradation Matrix (MDM) [3] and related Issue Management Tables (IMT) [4,5]. The PMDA and MDM have proven to be very complementary over the years. This approach is intended to develop a foundation for appropriate actions for keeping materials degradation from adversely impacting component integrity and safety and for identifying materials and locations where degradation can reasonably be expected in the future.

Extending reactor service to beyond 60 years will increase the demands on materials and components. Therefore, an early evaluation of the possible effects of extended lifetime is critical. The recent

NUREG/CR-7153 [2] gives a detailed assessment of many of the key issues in today's reactor fleet and provides a starting point for evaluating those degradation forms particularly important for consideration in extended lifetimes. While life beyond 60 will add additional time and neutron fluence, the primary impact will be increased susceptibility (although new mechanisms are possible).

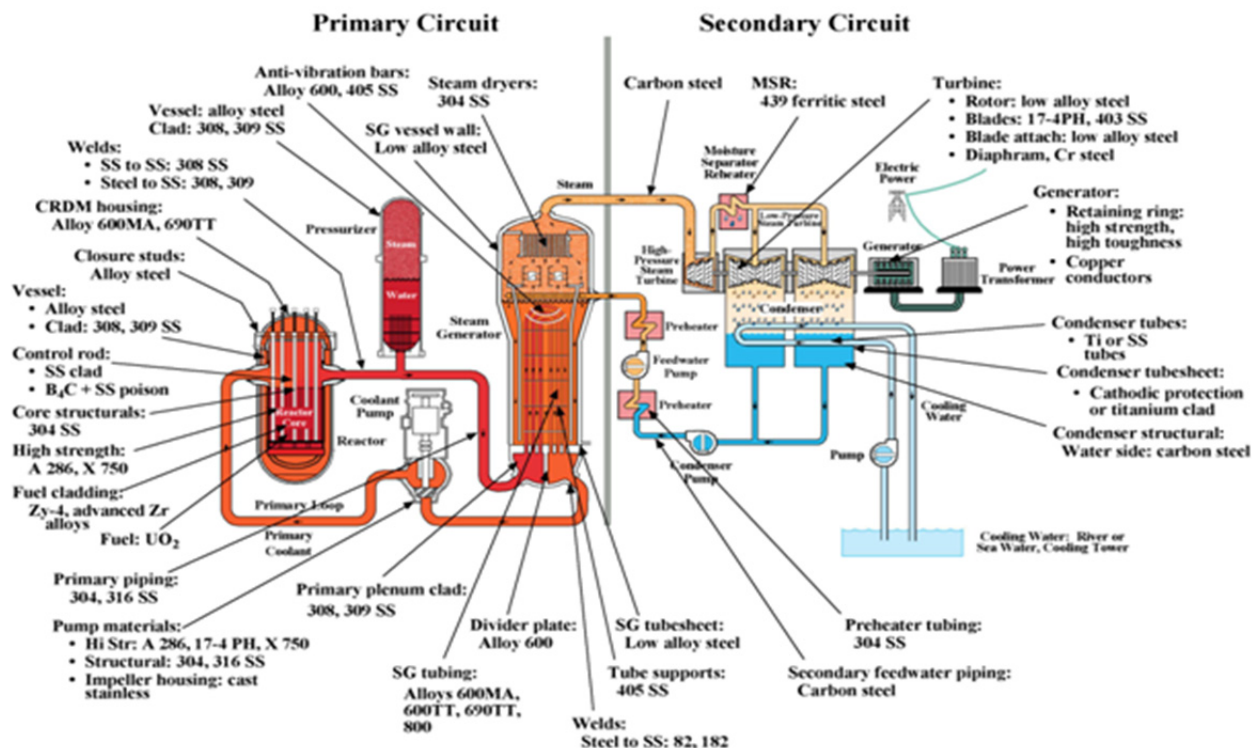


Figure 1: Sampling of the typical materials in a pressurized water reactor. Source: Staehle [1].

For RPVs, a number of significant issues have been recommended as deserving attention in future research activities. Large predictive uncertainties for embrittlement can result from limited data available at high fluences, for long times, and for alloy solute concentrations. The use of test reactors at high fluxes to obtain high fluence data is problematic for representation of the low flux conditions in reactor pressure vessels (RPVs). Late-blooming phases, especially for high-nickel welds, have been observed, and additional experimental data are needed in the high fluence regime where they are expected. Other discussed issues include specific needs regarding application of the fracture toughness master curve, data on long-term thermal aging, attenuation of embrittlement through the RPV wall, and development of an embrittlement trend curve based on fracture toughness.

For the reactor core and primary systems, several key areas have been identified. Thermo-mechanical considerations such as aging and fatigue must be examined. Irradiation-induced processes must also be considered for higher fluences, particularly the influence of radiation-induced segregation (RIS), swelling, and/or precipitation on embrittlement. Environment-induced degradation takes many forms in the primary reactor system, with stress corrosion cracking (SCC) of high interest for many components and irradiation-assisted SCC (IASCC) as a special case in the core region. Research in these areas can build

upon other ongoing programs in the light water reactor (LWR) industry as well as other reactor materials programs (such as fusion and fast reactors) to help resolve these issues for extended LWR life.

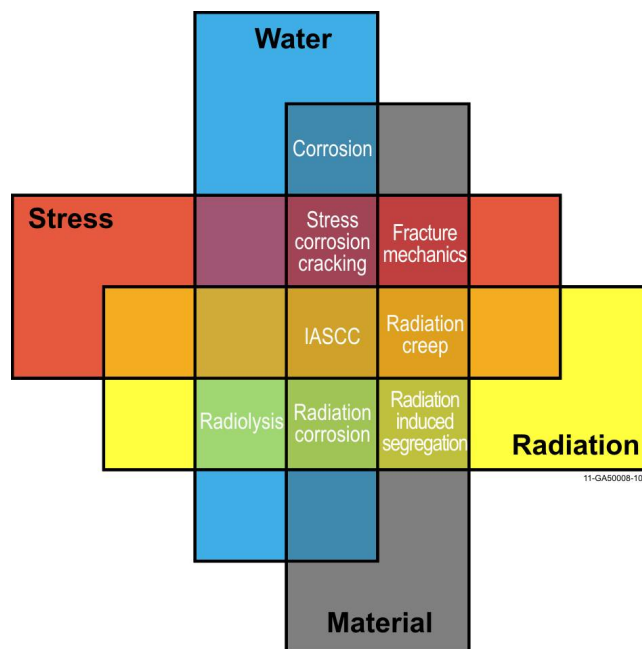
In the primary piping and secondary systems, corrosion is a key concern. Corrosion is a complex form of degradation that is strongly dependent on temperature, material condition, material composition, water purity, water pH, water impurities, and gas concentrations. The operating corrosion mechanism will vary from location to location within the reactor core, and a number of different mechanisms may be operating at the same time. These may include general corrosion mechanisms such as uniform corrosion, boric acid corrosion (BAC), flow-accelerated corrosion (FAC), and/or erosion corrosion, which will occur over a reasonably large area of material in a fairly homogenous manner. Localized corrosion modes occur over much smaller areas but at much higher rates than general corrosion and include crevice corrosion, pitting, galvanic corrosion, and microbiologically influenced (MBI) corrosion. Finally, environmentally assisted cracking (EAC) includes other forms of degradation that are closely related to localized or general corrosion with the added contribution of stress. In a LWR, a number of different environmentally assisted cracking mechanisms are observed: intergranular stress corrosion cracking (IGSCC), transgranular stress corrosion cracking (TGSCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and low temperature crack propagation (LTCP). Understanding the various modes of corrosion and identifying mitigation strategies is an important step for long-term service.

Fatigue damage from mechanical and/or environmental factors is the number one cause of failure in metallic components and has affected many different systems in service experience. The effects of environment on the fatigue resistance of materials used in operating pressurized water reactor (PWR) and boiling water reactor (BWR) plants are uncertain. There is a need to assess the current state of knowledge in environmentally assisted fatigue of materials in LWRs under extended service conditions.

In the area of welding technology, two critical long-standing welding-related technical challenges require further research and development (R&D), both fundamental and applied. The first is the need for an advanced weld simulation tool to support component life extension and reliable lifetime prediction, especially as related to the issue of residual stresses as a primary driving force for SCC. The second challenge is the development of new welding technologies for reactor repair and upgrade.

Concrete structures can also suffer undesirable changes with time because of improper specifications, a violation of specifications, or adverse performance of the cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete service life. Research is needed in a number of areas to ensure the long-term integrity of the reactor concrete structures. For example, radiation effects on containment concrete emerged as the most important degradation mechanism, mainly driven by insufficient data to improve the level of knowledge about the effects of irradiation on concrete mechanical properties. Alkali-aggregate reaction (AAR), acid attack and creep emerged as secondarily important mechanisms. The biggest surprise in this analysis is the result that susceptibility to fracture emerged as the least important mechanism. This should be interpreted to apply only to concrete cracking of the generally known type that is accounted for in the structural design.

Clearly, the demanding environments of an operating nuclear reactor may impact the ability of a broad range of materials to perform their intended function over extended service periods. Routine surveillance and repair/replacement activities can mitigate the impact of this degradation; however, failures still occur. With reactors being licensed to operate for periods up to 60 years or beyond and power uprates being planned, many of the plant SSCs will be expected to tolerate more demanding environments for longer periods. The longer plant operating lifetimes may increase the susceptibility of different SSCs to degradation and may introduce new degradation modes. For example, in the area of crack-growth mechanisms for Ni-base alloys alone, there are up to 40 variables known to have a measurable effect. Further, many variables have complex interactions (Figure 2 [6]). In this same instance (crack-growth mechanisms for Ni-base alloys), a purely experimental approach would require greater than a trillion experiments to address the variables and interactions. Therefore, the application of modern materials science will be necessary to resolve these issues.



4

In the past two decades, there have been great gains in techniques and methodologies that can be applied to the nuclear materials problems of today. Indeed, modern materials science tools (such as advanced characterization and computational tools) must be employed. Furthermore, due to the complex nature of these degradation modes and the synergistic effects between them, combined approaches must be taken. Materials research must include a mix of experimental testing performed in simulated reactor environments under accelerated conditions, the examination of harvested components that experienced actual service conditions over long periods of time, and the modeling or simulating of the degradation effects. Scientific research within the Materials Aging and Degradation Pathway approach multiple tasks within the framework shown in Figure 3. Individual research thrusts within the pathway provide contributions to the overall pathway goal through high quality scientific measurement of materials performance to understand the active modes and mechanism of degradation. This is through combinations of research experimentation, modeling or simulation, and information obtained from in-service exposed materials. The interdependence of the three research methods is important to understand, as modeling provides the ability to evaluate materials behavior subjected to a large variability of inputs that would make experimental testing costly and time consuming. However, models require validation through either harvested material examination or experimental testing. Similarly, accelerated irradiation testing is necessary to understand high fluence behavior, but must be judged based on available harvested or modeling simulation to assess the impact of flux dependent forms of materials degradation.

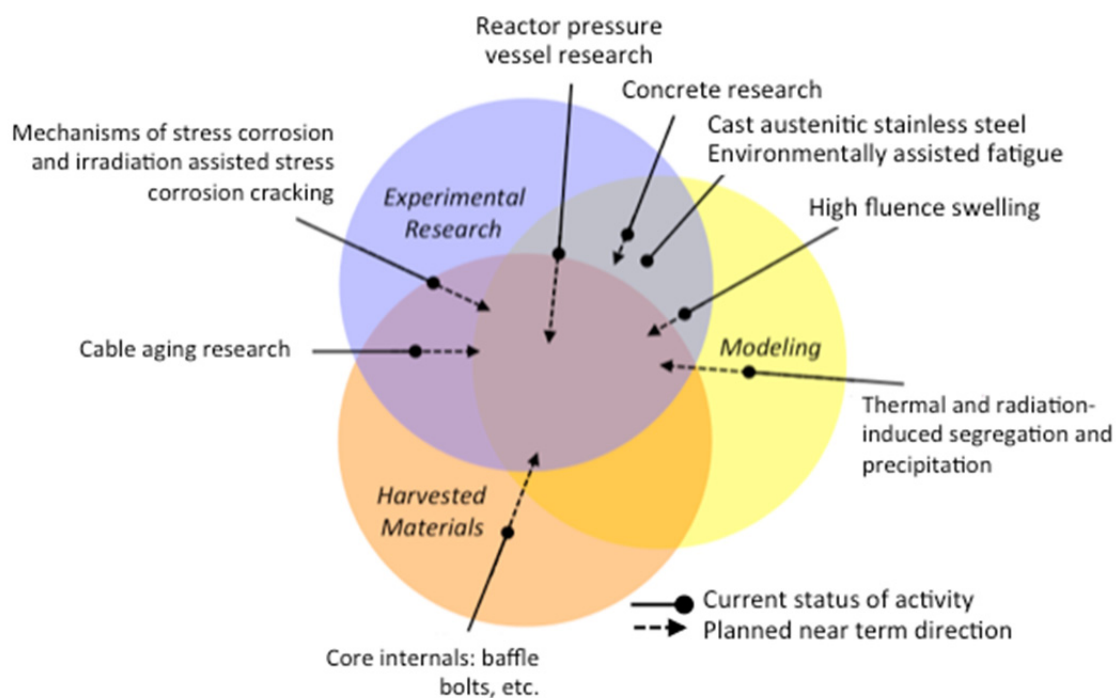


Figure 3: Methodology utilized in addressing the complex research needs within the Materials Aging and Degradation Pathway. Examples of some of the main research areas are shown, with current and near term activities shown.

While specific tools and the science-based approach can be described in detail for each particular degradation mode, many of the diverse technical topics and information needs in this area can be organized into a few key areas. These could include mechanisms of materials degradation, mitigation strategies, and modeling and simulation. While all components (except perhaps the RPV) can be replaced, decisions to simply replace components may not be economically favorable. Therefore, understanding,

controlling, and mitigating materials degradation processes and establishing a sound technical basis for long-range planning of necessary replacements are key priorities for extended reactor operations and power uprate considerations.

As noted above, there are many forms of materials degradation in a nuclear power reactor. Many of these are highly dependent upon a number of different variables, creating a complex scenario for evaluating lifetime extensions. Nonetheless, many of the diverse topics and needs described earlier can be organized into a few research thrust areas. These could include measurements and mechanisms of degradation, mitigation strategies, modeling and simulations, monitoring, and management.

Measurements of degradation: High-resolution measurements of degradation in all components and materials are essential to assess the extent of degradation under extended service conditions, support development of mechanistic understanding, and validate predictive models. High quality data are of value to regulatory and industry interests in addition to academia.

Mechanisms of degradation: Basic research to understand the underlying mechanisms of selected degradation modes can lead to better prediction and mitigation. For example, research on IASCC and PWSCC would be very beneficial for extended lifetimes and could build on other existing programs within EPRI and NRC.

Modeling and simulation: Improved modeling and simulation efforts have great potential in reducing the experimental burden for life extension studies. These methods can help interpolate and extrapolate data trends for extended life. Simulations predicting phase transformations, radiation embrittlement, and swelling over component lifetimes would be extremely beneficial to licensing and regulation in extended service.

Monitoring: While understanding and predicting failures are extremely valuable tools for the management of reactor components, these tools must be supplements to active monitoring. Improved monitoring techniques will help characterize degradation of core components. For example, improved crack detection techniques will be invaluable. New nondestructive examination techniques may also permit new means of monitoring pressure vessel embrittlement or swelling of core internals.

Mitigation strategies: While some forms of degradation have been well researched, there are few options in mitigating their effects. Techniques such as post-irradiation annealing have been demonstrated to be very effective in reducing hardening of entire pressure vessels. Annealing may be effective in mitigating IASCC, based on initial studies. Water chemistry techniques such as NobelChem have been very effective in reducing some corrosion problems. Additional research in these areas may provide other alternatives to component replacement.

The Light Water Reactor Sustainability (LWRS) program is designed to support the long-term operation of existing domestic nuclear power generation with targeted collaborative research programs into areas beyond current short-term optimization opportunities [7]. Within the LWRS program, four pathways have been initiated to perform research essential to informing relicensing decisions. The Materials Aging and Degradation (MAaD) pathway within the LWRS program is charged with the development of the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in reactors. The work will provide data and methods to assess performance of SSCs essential to

safe and sustained reactor operations. The R&D developed in this program will be used by utilities, industry groups, and regulators to affirm and define operational and regulatory requirements and limits for materials subject to long-term operation conditions.

2. Research and Development Purpose and Goals

Materials research provides an important foundation for licensing and managing the long-term, safe, and economical operation of nuclear power plants. Aging mechanisms and their influence on nuclear power plant SSCs are predictable with sufficient confidence to support planning, investment, and licensing for necessary component repair, replacement, and relicensing. Understanding, controlling, and mitigating materials degradation processes are key priorities. While our knowledge of degradation and surveillance techniques are vastly improved, unexpected degradation can still occur. Proactive management is essential to help ensure that any degradation from long-term operation of nuclear power plants does not affect the public's confidence in the safety and reliability of those nuclear power plants.

The strategic goals of the MAaD pathway are to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants and to provide data and methods to assess performance of SSCs essential to safe and economically sustainable nuclear power plant operations.

The US DOE (through the MAaD pathway) is involved in this R&D activity for the following reasons:

1. MAaD tasks provide fundamental understanding and mechanistic knowledge via science-based research. Mechanistic studies provide better foundations for prediction tool development and focused mitigation solutions. These studies also are complementary to industry efforts to gain relevant, operational data. The US national laboratory and university systems are uniquely suited to provide this information given their extensive facilities, research experience, and specific expertise. Specific outcomes of these fundamental tasks include mechanistic understanding of key degradation modes, elucidating the role of composition, material history, and environment in degradation. In many of these tasks, models to predict susceptibility over a lifetime will be developed. In some tasks, understanding if a mode of degradation is a true concern is a key outcome.
2. While understanding and predicting failures are extremely valuable tools for the management of reactor components, active monitoring of materials degradation and alternatives to component replacement are also invaluable. Improved monitoring techniques will help characterize degradation of core components. Selected MAaD tasks are focused on the development of high-risk, high-reward technologies to understand, mitigate, or overcome materials degradation. This type of alternative technology research is uniquely suited for government roles and facilities. These pursuits are also outside the area of normal interest for industry sponsors due to risk of failure. New nondestructive examination techniques may permit a means of monitoring components such as the RPV, core internals, cables, or concrete. Specific mitigation research tasks in this area include development of advanced welding techniques and annealing processes to overcome component damage. Specific outcomes of these tasks will be the transfer of advanced methodologies to industry.

3. MAaD tasks support collaborative research with industry and/or regulators (and meet at least one of the objectives above). The focus of these tasks is on supporting and extending industry capability by providing expertise, unique facilities, or fundamental knowledge.

Combined, these thrusts provide high quality measurements of degradation modes, improved mechanistic understanding of key degradation modes, and predictive modeling capability with sufficient experimental data to validate these tools; new methods of monitoring degradation; and development of advanced mitigation techniques to provide improved performance, reliability, and economics.

This information must be provided in a timely manner in order to support license renewal decisions, which will be made by the first utilities in the coming years. Key research outcomes will be required before 2020. Near-term research is focused primarily on providing mechanistic understanding, predictive capabilities, and high-quality data. All three of these outputs will inform decisions and processes by both industry and regulators. Longer-term research will focus on alternative technologies to overcome or mitigate degradation. The high-priority tasks initiated in the past five years have all addressed key issues. The diversity of the research thrusts is shown in Figure 4. All areas of the plant are being addressed. Further, task outputs and products are being designed to inform relicensing decisions and regulatory processes and impacts, as will be discussed in detail in sections to follow.

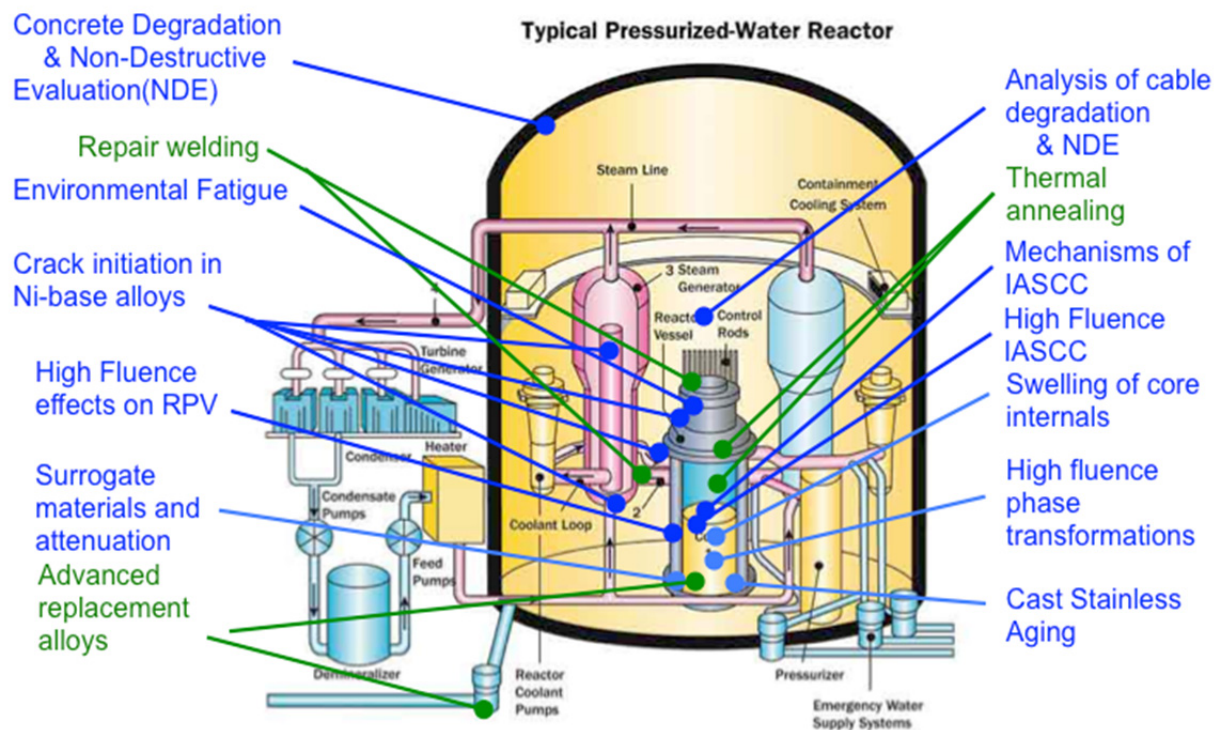


Figure 4: Research tasks supported within MAaD pathway of the Light Water Reactor Sustainability program.

3. Materials Aging and Degradation Pathway Research and Development Areas

As noted in Chapter 1, materials aging and degradation is complex in a modern nuclear power plant and involves many different classes of materials in very diverse environments. The goals of the MAaD pathway are to help prioritize these diverse materials degradation issues, to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants, and to provide data and methods to assess performance of SSCs essential to safe and economically sustainable nuclear power plant operations.

The MAaD pathway activities were originally organized into six principal areas: (1) reactor metals, (2) concrete, (3) cables, (4) buried piping, (5) mitigation strategies, and (6) integrated research activities with industry, universities, and across LWRs pathways. Each of these primary topics consists of multiple research projects within the pathway. Over the last two years, research into buried piping has been deferred as the nuclear industry has significant programs ongoing in this topical area. The LWRs program continues to evaluate this area for gaps and needs relative to extended service. These research areas cover material degradation in SSCs that were designed for service without replacement throughout the life of the plant. Management of long-term operation of these components can be difficult and expensive. As power plant licensees seek approval for extended operation, the way in which these materials age beyond 60 years will need to be evaluated and their capabilities reassessed to ensure that they maintain the ability to perform their intended functions in a safe and reliable manner. There are additional activities to support management of the MAaD, a systematic characterization of degradation modes, and unique integration activities with other LWRs pathways and industry.

This section first provides a discussion on the rationale for the selection of research tasks within the MAaD pathway. Each major research area is summarized, including a detailed description of all ongoing and planned research tasks. In the description for each work package, the specific workscope is given along with the expected outcomes. Key deliverables are also listed with the expected value for key stakeholders for several of the highest-level milestones.

3.1 Identification and Prioritization of Research Activities

Given the diversity of materials, environments, and histories noted above, there are many competing needs for research that must be addressed in a timely manner to support relicensing decisions. To meet the programmatic goals and support DOE mission requirements, research tasks within the MAaD pathway must meet at least one of five key criteria:

- Degradation modes that are already occurring and will grow more severe during extended lifetimes
- Degradation modes for which there is little or no mechanistic understanding and for which long-term research is needed
- Degradation modes for which there is little or no supporting data and that may be problematic for extended lifetimes

- Degradation modes for which follow-on work can complement other national or international efforts
- Areas for which technical progress can be made in the near term.

Identifying, formulating, and prioritizing all of these competing needs has been done in a collaborative manner with industrial and regulatory partners. The primary objective of a workshop focusing on materials aging and degradation, held at the EPRI offices in Charlotte, North Carolina, on August 5 and 6, 2008, was to identify an initial list of the most pressing research tasks. Twenty technical experts, providing broad institutional representation, attended the MAaD pathway workshop. Three national laboratories, two universities, two nuclear reactor vendors, a nuclear power plant utility, and nine key experts from EPRI participated in the discussions. Technical backgrounds and expertise included radiation effects; corrosion and stress corrosion cracking (SCC); water chemistry effects; predictive modeling; aging; and high-temperature design methodology covering RPVs, core internals, cabling, concrete, piping, and steam generators.

Points of discussion included organization and structure of the MAaD pathway, need and benefits of an advisory group, and identification and prioritization of research tasks to support the LWRS program. Workshop participants identified a total of 47 different research tasks to be considered. This number was quickly reduced to 39 tasks by combining similar needs and eliminating overlapping efforts. Each of these tasks met one of the criteria described above to ensure relevance to this pathway and the LWRS program strategic goals.

All of the 39 tasks that were identified were believed to be relevant to the LWRS program and important to life extension decisions. However, the technical need was not equal for each of the tasks. Therefore, every task was classified as high, medium, or low priority. When considering task prioritization, workshop participants determined that degradation modes that could influence the primary pressure boundary or core structural integrity (including the core internal structures, RPV, and primary piping) were all high-priority tasks because of the negative outcomes associated with such a failure. Also, modes of degradation that were unknown or modes of degradation in components that could not be accessed or replaced (e.g., concrete structures) were designated as high priority. Of the original 39 tasks, 13 were considered high priority, 22 were considered medium priority, and 4 were considered low priority. The 13 high-priority tasks were considered for initiation in FY 2009.

In a separate exercise, each participant was polled on the modes of degradation they felt were the most problematic for long-term reactor operation (Figure 5). Almost every participant identified potential embrittlement of RPV steels and IASCC of core internals as a key concern. Also of high importance was SCC of Ni-base alloys and austenitic steels in the primary water loop. These trends match the input presented above.

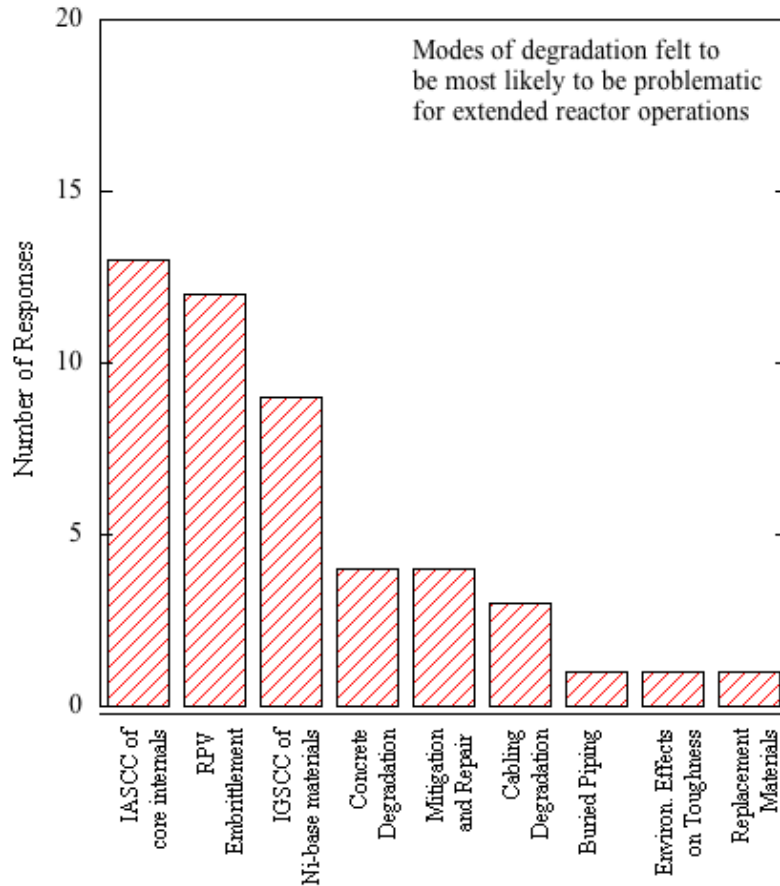


Figure 5: Summary of modes of degradation that are the most likely to be problematic for long-term operation of nuclear reactor power plants.

Since FY 2009, additional tasks from this list have been pursued. Research has identified additional needs, and these research topics have also been considered. Continued dialogue with EPRI, NRC, vendors, utilities, and other institutions around the world has helped prioritize these emerging needs into the MAaD research portfolio. All research tasks are described in more detail below.

Ensuring that the research remains focused on closing the most important knowledge gaps remains a high priority within the Materials Aging and Degradation Pathway. In 2012, the LWRS program and US NRC staff recognized that an organized, Phenomena Identification Ranking Table (PIRT) approach to organizing materials degradation could be used to support the development of technical bases for subsequent license renewal. This activity included a series of expert panel deliberations and was termed the Expanded Proactive Materials Degradation Analysis (EMDA). EMDA represents a significant broadening of scope relative to PMDA [2]. First, the analytical timeframe is extended from 40 years to 80 years, encompassing the subsequent license renewal-operating period. Second, the materials and systems addressed in EMDA are generally extended to all of those which fall within the scope of aging management review for license renewal. Thus, in addition to piping and core internals, EMDA also includes the reactor pressure vessel (RPV), electrical cables, and concrete structures. A diverse expert panel was assembled for each of the four assessments. Each panel was composed of at least one member representing the regulator, industry [e. g., the Electric Power Research Institute (EPRI), vendors], the U.S.

national laboratories, academia, and an international aging degradation expert. The final findings of these expert panels will be publicly released in 2014 and will be used to continue to prioritize research and address knowledge gaps for life extension decisions.

3.2 Management Activities

There are two key activities supporting management of the MAaD pathway. While these activities do not directly produce measurements, mechanisms, or models, they are essential in ensuring that research is performed in an efficient manner and in developing key partnerships and relations. In addition, efforts in this pathway area help determine and prioritize research tasks. The Project Management and Assessment and Integration tasks support these activities, respectively. Both are described in more detail below.

The Project Management task is designed to support routine project management activities and new program development tasks, report generation, travel, meetings, and benchmarking. In addition, this pathway task is essential to support the integrated and coordinated effort that is required to successfully identify and resolve materials degradation issues. A key outcome of this task is the annual development of a research plan and coordination with other stakeholders. In addition, this task is charged with support updates to the LWRS Integrated Program Plan.

Another key objective of this research task is to provide a comprehensive assessment of materials degradation, relate to consequences of SSCs and economically important components, incorporate results, guide future testing, and integrate with other pathways and programs. This task will provide an organized and updated assessment of key materials aging degradation issues and support NRC and EPRI efforts to maintain an updated PMDA or MDM documents. Successful completion will provide a valuable means of task identification and prioritization within this pathway, as well as identify new needs for research.

In previous years, an expanded materials degradation analysis (EMDA) of degradation mechanisms for 60–80 years or beyond was identified as a useful tool in further prioritizing degradation for research needs. However, expansion of the original PMDA to longer timeframes and additional SSCs is a large undertaking. Therefore, via joint discussions between DOE and NRC, it was decided that the EMDA would consist of separate and focused documents covering the key SSCs. This would yield a series of independent assessments that, when combined, would create a comprehensive EMDA. Four separate assessments were developed:

- core internals and primary and secondary piping (or current materials in NUREG/CR-6923 [2]),
- reactor pressure vessels,
- concrete civil structures, and
- electrical power and instrumentation and control (I&C) cabling and insulation.

Each separate assessment has chartered an expert group with research, regulatory, and industry perspectives. These expert panels were charged with providing an analysis of key degradation modes for current and expected future service, key degradation modes expected for extended service, and suggested research needs to support extended operation in the subsequent renewal periods (i.e., 60–80 years). This valuable resource was delivered in 2014 [2] and is being used as a prioritization tool within LWRS today.

Current efforts continue to evaluate the research tasks within the pathway to ensure that the key degradation issues and primary materials systems of concern from the EMDA activities are being appropriately addressed. This is through routine communication and workshop meetings to discuss technical roadmaps from which the LWRS and its collaborators follow to reach the primary goals of the program towards assessment of long-term materials performance, condition monitoring and mitigation efforts to maintain energy production through nuclear power. The management task also engages discussions with industry and operators to keep abreast of immediate or emerging materials issues and to provide transfer of technical knowledge from pathway research efforts.

Products: Coordinated research management on a continuing basis

Lead Organization: Oak Ridge National Laboratory (ORNL)

Current Partners: NA

Project Milestones/Deliverables:

- Provide updated Plan for the MAaD pathway, *on annual basis*
- Provide updated MAaD pathway input to the LWRS Integrated Program Plan, *on annual basis*

Value of Key Milestones to Stakeholders: Delivery of the final EMDA in NUREG form was completed 2014 and is expected to have lasting value to all stakeholders. The LWRS program has used this as a tool for identifying and prioritizing research.

3.3 Reactor Metals

As described above, numerous types of metal alloys can be found throughout the primary and secondary systems of reactors. Some of these materials (in particular, the reactor internals) are exposed to high temperatures, water, and neutron flux. This challenging operating environment creates degradation mechanisms in the materials that are unique to reactor service. Research programs in this area will provide a technical foundation to establish the ability of those metals to support nuclear reactor operations to 60 years and beyond. The nine primary activities for this area are listed below along with key outcomes for each task.

- Mechanisms of IASCC in stainless steels: provides understanding of role of composition, history, and environment on IASCC and model capability.
- High-fluence effects on RPV steels: provides evaluation of risk for high fluence embrittlement after long service life; mechanistic understanding of effects of fluence, flux, and composition on hardening; and model capability. This task also evaluates the viability of miniature fracture toughness testing of irradiated materials to provide further scientific information to surveillance materials.

- SCC initiation in Ni-base alloys: provides mechanistic understanding of precursor states on crack initiation to develop strategies for mitigation.
- High-fluence effects on IASCC of stainless steels: provides evaluation of new factors at high fluence, first data for high fluence conditions, and validation of models and mechanisms.
- Evaluation of swelling effects in high-fluence core internals: provides evaluation of risk for high-fluence core internal components to swelling and development of a predictive model capability.
- Evaluation of irradiation-induced phase transformations in high-fluence core internals: provides evaluation of risk for high-fluence core internal components and RPV steels to embrittlement due to phase transformations and development of a predictive model capability.
- Material variability and attenuation effects on RPV steels: provides mechanistic information on attenuation effects through RPV wall thickness, validation of high-flux irradiations for surveillance capsules, alternative monitoring concepts, and validation of models.
- Environmental fatigue: provides mechanistic understanding of key variables in environmental fatigue to develop strategies for management.
- Thermal aging of cast austenitic stainless steel: provides evaluation of the effects of long-term thermal aging of cast austenitic stainless steel, through accelerated thermal aging tests supported by thermodynamic modeling of phase development that impacts aged mechanical properties.
- Thermodynamic tools for evaluation of radiation effects: provides the development of computational tools by coupling Radiation Induced Segregation (RIS) model with computational thermodynamics for simulation of RIS and Radiation Induced precipitation (RIP) in the steels used in Light Water Reactors.

3.3.1 High Fluence Effects on Reactor Pressure Vessel Steels

The last few decades have seen remarkable progress in developing a mechanistic understanding of irradiation embrittlement for the RPV. This understanding has been exploited in formulating robust, physically based, and statistically calibrated models of Charpy V-notch (CVN)-indexed transition temperature shifts. However, these models and our present understanding of radiation damage are not fully quantitative and do not treat all potentially significant variables and issues.

Similarly, developments in fracture mechanics have led to a number of consensus standards and codes for determining the fracture toughness parameters needed for development of databases that are useful for statistical analysis and establishment of uncertainties. The CVN toughness is a qualitative measure that must be correlated with the fracture toughness and crack-arrest toughness properties necessary for structural integrity evaluations. Direct measurements of the fracture toughness properties are desirable to reduce the uncertainties associated with correlations.

The progress notwithstanding, however, there are still significant technical issues that need to be addressed to reduce the uncertainties in regulatory application. The issues regarding irradiation effects, briefly summarized in this section, are those identified by a cross section of researchers in the international community. Of the many significant issues discussed, those deemed to have the most impact on the current regulatory process and life extension are listed below and include both experimental and modeling needs. Moreover, the combination of irradiation experiments with modeling and microstructural studies provides an essential element in aging evaluations of RPVs.

Limited data at high fluences, for long times, and for specific alloy chemistry create large uncertainties for embrittlement predictions. This issue is directly related to life extension with the number of plants requesting license extension to 60 years and those expected to request 80 years. Simply stated, extending operation from 40 years to 80 years will double the neutron exposure for the RPV. Moreover, because the recent pressurized thermal shock (PTS) reevaluation project has resulted in lower average failure probabilities for PWRs, many plants are increasing their operating power level, which will further increase the fluence. To obtain data at the high fluences for life extension will require the use of test reactor experiments that use high neutron fluxes. Substantial research is needed to enable application of data obtained at high flux to RPV conditions of low flux and high fluence. Furthermore, an improved understanding of the precipitate development that occurs in RPV steels over time and the effect that alloy chemistry has on long-term properties is needed. Mechanical properties of the RPV steel at high fluence is dependent on the contribution of the so-called late blooming phases in the form of Mn-Ni-Si precipitates which occur in both Cu-bearing and nearly Cu-free concentrations. An example of the influence alloy composition has on hardening levels are given in Figure 6. The importance of understanding the role of alloy composition, flux and total fluence is important, as current regulatory models including both the Eason-Odette- Nanstad-Yamamoto (EONY) model and the new American Society for Testing and Materials (ASTM) E900 Standard can significantly under predict hardening in steels at high fluence levels.

The objective of this research task is to examine and understand the influence of irradiation at high fluences on RPV embrittlement. Irradiation of RPV steels may cause embrittlement of the primary containment structure. Both surveillance capsule data and single-variable experiments may be required to evaluate potential for embrittlement and to provide a better mechanistic understanding of this form of degradation. Acquisition of samples from past programmatic campaigns (such as NRC programs), specimens harvested from decommissioned reactors, surveillance specimens from operating nuclear power plants, and materials irradiated in new test campaigns all have value in understanding high fluence effects. Testing will include impact and fracture toughness evaluations, hardness, and microstructural analysis (atom probe tomography, small angle neutron scattering, and/or positron-annihilation spectroscopy). These research tasks all support development of a predictive model for transition-temperature shifts for RPV steels under a variety of conditions. This tool can be used to predict RPV embrittlement over a variety of conditions key to irradiation-induced changes (e.g., time, temperature, composition, flux, and fluence) and extends the current tools for RPV management and regulation to extended-service conditions. This model will be delivered in 2018 in a detailed report, along with all supporting research data. In addition, the library of assembled materials will be available for examination and testing by other stakeholders.

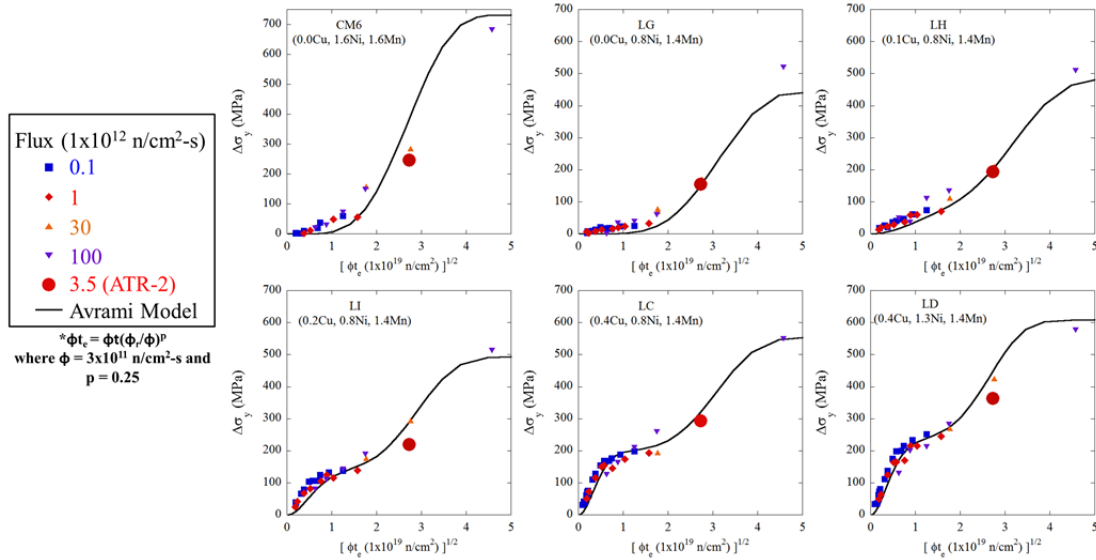


Figure 6: Preliminary University of California Santa Barbara (UCSB) Avrami model predicting $\Delta\sigma_y$ as a function of fluence for the core UCSB split melt steels from previous experimental irradiation programs. [8].

Current work within this research area also includes the evaluation of miniature compact tension (MCT) fracture toughness specimens that can be machined from the halves of tested CVN impact bars. The CVN bars are commonly used specimen geometry for surveillance programs, but only provide a qualitative measure of mechanical properties. The testing of MCT's from Charpy specimens will allow the determination and monitoring of actual fracture toughness instead of indirect predictions using CVN specimens. Furthermore, multiple MCT's can be fabricated from a single Charpy specimen. This effort will validate fracture toughness data derived from MCT's with previously characterized specimens towards the modification of standard E1921 to develop a master curve that accommodates MCT's.

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: ORNL with support from the University of California–Santa Barbara (UCSB) and the University of Wisconsin

Current Partners: Constellation Energy (RPV surveillance coupons), Westinghouse (RPV sample material), Vattenfall (RPV sample material), Nuclear Scientific User Facility (grant for irradiation campaign via UCSB)

Project Milestones/Deliverables

- Provide report detailing testing, progress, and results for high-fluence RPV analysis, *on annual basis*
- Acquire industry-relevant RPV specimens from nuclear power plant, *July 2011 – COMPLETED*

- Complete detailed analysis of RPV samples from nuclear power plant, November 2012 - COMPLETED
- Initiate post-irradiation examination of newly irradiated RPV specimens from ATR campaign, *September 2015*
- Complete acquisition of experimental data on commercial and model RPV alloys, September 2014: *COMPLETED*
- **Provide model for transition temperature shifts in RPV steels, *September 2017***
- **Establish validation of model for transition temperature shifts in RPV steels, *February 2018***
- Future milestones and specific subtasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Both industry and regulators will use the experimental data and model tools. Completion of data acquisition to permit prediction of embrittlement in RPV steels at high fluence is a major step in informing life-extension decisions, and high-quality data can be used to inform operational decisions for the RPV by industry. For example, data and trends will be essential in determining operating limits. The data will also allow for extension of regulatory limits and guidelines to extended service conditions. The delivery of a validated model for prediction of transition temperature shifts in RPV steels in 2017 will allow for estimation of RPV performance over a wide range of conditions. This will enable extension of current tools for RPV embrittlement (e.g., FAVOR) to extended service conditions.

3.3.2 Material Variability and Attenuation Effects on Reactor Pressure Vessel Steels

The subject of material variability has experienced increasing attention in recent years as additional research programs have begun to focus on the development of statistically viable databases. With the development of the Master Curve approach for fracture toughness and the potential use of elastic-plastic fracture-toughness data for direct application to the RPV, attention has focused on the issue of material variability. Many surveillance programs contain CVN specimens of a different heat of base metal or a different weld than that in the RPV. This issue has received attention within the industry and is under evaluation by the NRC. Application of the Master Curve methodology to RPVs is not likely to occur without resolution of this issue, including development and acceptance of the associated uncertainties.

Further, there is still some controversy over the way in which embrittlement variations through the RPV wall arising from attenuation of the neutron flux should be estimated. The current methodology is based on neutron fluence greater than 1 MeV, but the use of displacements per atom (dpa) is more technically sound. Several types of research are needed to better resolve both the issue of the proper dose unit and to provide a proper framework for assessing attenuation. Development of the attenuation model can be accomplished through test reactor experiments (such as that recently sponsored by the International Atomic Energy Agency in a Russian test reactor) or through direct examination of a decommissioned RPV such as that of the Zion nuclear power plant.

The objectives of this task involve developing new methods to generate meaningful data out of previously tested specimens. Embrittlement margins for a vessel can be accurately calculated with supplementary

alloys and experiments such as higher flux test reactors. The potential for non-conservative estimates resulting from these methodologies must be evaluated to fully understand the potential influence on safety margins. Critical assessments and benchmark experiments will be conducted. Harvesting of through-thickness RPV specimens may be used to evaluate attenuation effects in a detailed and meaningful manner. As above, testing will include impact and fracture toughness evaluations, hardness, and microstructural analysis (atom probe tomography, small angle neutron scattering, and/or positron-annihilation spectroscopy). The results of these examinations can be used to assess the operational implications of high-fluence effects on the RPV. Furthermore, the predictive capability developed in earlier tasks will be modified to address these effects.

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: ORNL with support from the University of California–Santa Barbara

Current Partners: Constellation Energy (RPV surveillance coupons), Westinghouse (RPV sample material), Vattenfall (RPV sample material)

Project Milestones/Deliverables

- Provide annual report detailing year’s testing, progress, and results, *on annual basis*
- Complete plan for attenuation and material variability studies evaluation, *September 2012 – COMPLETED*
- Complete a detailed review of the NRC PTS reevaluation project relative to the subject of material variability and identify specific remaining issues, *December 2018*
- Complete analysis of copper variations through-RPV thickness and evaluate uncertainties with regard to irradiation-induced degradation and safety margins, *March 2019*
- **Complete analysis of hardening and embrittlement through the RPV thickness for the Zion RPV sections, *September 2020***
- Future milestones and specific subtasks will be based on the results of the previous year’s testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: The analysis of hardening and variability through the thickness of an actual RPV section from service has considerable value to all stakeholders. This data will provide a first look at embrittlement trends through the thickness of the RPV wall and inform operating limits, fracture mechanics models, and safety margins.

3.3.3 Mechanisms of Irradiation-Assisted Stress Corrosion Cracking

Over the forty-year lifetime of a light water reactor, internal structural components may expect to see up to $\sim 10^{22}$ n/cm² in a BWR and $\sim 10^{23}$ n/cm² in a PWR ($E > 1$ MeV), corresponding to ~ 7 dpa and 70 dpa, respectively. Extending the service life of a reactor will increase the total neutron fluence to each component. Fortunately, radiation effects in stainless steels (the most common core constituent) are also the most examined as these materials are also of interest in fast-spectrum fission and fusion reactors where higher fluences are encountered.

In addition to elevated temperatures, intense neutron fields and stress components must also be able to withstand a corrosive environment. Temperatures typically range from 288°C in a BWR up to 360°C in a PWR (in some locations with high gamma heating) although other water chemistry variables differ more significantly between the BWRs and PWRs. While all forms of corrosion are important in managing a nuclear reactor, IASCC has received considerable attention over the last four decades due both to its severity and unpredictability. IASCC affects core internal structures, and sudden failures to safety components could be catastrophic. The combined effects of corrosion and irradiation create potential for increased failures due to IASCC. Over the last two years, the University of Michigan has developed new testing techniques to permit examination of the early stages of crack initiation. Post test characterization efforts at ORNL has also yielded new insights on the role of defect to defect interactions that can create stress risers as well as understanding the effect of grain boundary orientation relative to the state of stress and the conditions which favor the promotion of crack nucleation and growth, as shown in Figure 7. This is essential to help provide insights and data required for predictive capability and ultimately mitigation of this form of degradation.

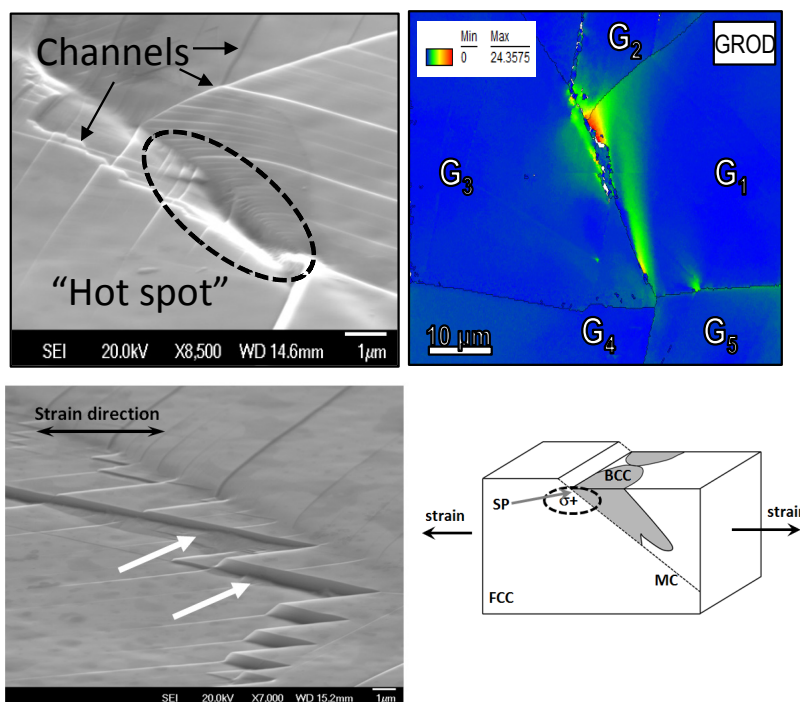


Figure 7: IASCC susceptibility observed in 304 stainless steel irradiated to 4.4 dpa and tested in a bend test at different stress levels to determine interaction of dislocation channels and their impact on crack initiation in recent tests in LWRS program. The conversance of the channels at the grain boundary (top left) produces a high degree of localized strain as identified in the Grain Reference Orientation Deviation (GROD) image (top right). Further investigations have revealed the formation of strain induced martensite domains at the high locally strained regions. Source: ORNL [9].

Despite over thirty years of international study, the underlying mechanism of IASCC is still unknown. More recent work led by groups such as the Cooperative IASCC Research Group has identified other

possible causes that are currently being investigated as possible drivers for IASCC. Further work at the University of Michigan has begun looking at the effect of water chemistry on the crack growth behavior of irradiate stainless steel.

The objective of this work is to evaluate the response and mechanisms of IASCC in austenitic stainless steels with single-variable experiments. Crack growth rate tests and complementary microstructure analysis will provide a more complete understanding of IASCC by building on past EPRI-led work for the Cooperative IASCC Research Group. Experimental research will include crack-growth testing on high-fluence specimens of single-variable alloys in simulated LWR environments, tensile testing, hardness testing, microstructural and microchemical analysis, and detailed efforts to characterize localized deformation. Combined, these single variable experiments will provide mechanistic understanding that can be used to identify key operational variables to mitigate or control IASCC, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, in the long-range, design IASCC-resistant materials.

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: University of Michigan, with support from ORNL

Current Partners: EPRI (cost-sharing and technical input)

Project Milestones/Deliverables

- Complete report on testing progress to determine mechanisms of IASCC, *on annual basis*
- Procure other commercial materials of interest for testing of IASCC response, *December 2012 - COMPLETED*
- Initiate IASCC-susceptibility evaluation on supplementary specimens and conditions, *March 2013 - COMPLETED*
- Complete mechanistic testing for IASCC research, *December 2014-COMPLETED*
- Initiate predictive modeling and theoretical studies to develop predictive capability for IASCC under extended service conditions of core internal components, *March 2015*
- Initiate benchmarking testing for IASCC predictions using plant component materials, *March 2017*
- **Deliver predictive model capability for IASCC susceptibility, March 2019**
- Detailed testing and specific subtasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing research to identify the mechanisms of IASCC (2014) is an essential step to predicting the extent of this form of degradation under extended service conditions. Understanding the mechanism of IASCC will enable more focused material inspections, material replacements, and more detailed regulatory guidelines. In the long-term, mechanistic understanding also enables development of a predictive model (2019), which has been sought for IASCC for decades.

3.3.4 High Fluence Irradiation-Assisted Stress Corrosion Cracking

As noted above, IASCC in 304 and 316 stainless steel is expected to become more severe or to influence previously resistant alloys. Long-term service will result in very high accumulations of radiation damage. Unfortunately, very little IASCC or fracture toughness data exists for high-fluence specimens of austenitic stainless steels (wrought or cast) or weldments within the reactor core. The objectives of this task are to assess high-fluence effects on IASCC for core internals. Crack growth-rate testing is especially limited for high-fluence specimens. Intergranular fracture observed in recent experiments suggests more work is needed. Also of interest is identification of high-fluence materials available for research and testing in all tasks.

Current research involves fracture mechanics testing of the crack propagation in high fluence 304 stainless to address the effect of high fluence microstructural features on IASCC, such as radiation-induced swelling. This task is also involved in evaluating the efficiency of the IASCC crack growth rate mitigation in hydrogen water chemistry conditions, which might not be as effective at high fluence conditions as they are at low. The later work is in collaboration with the Japanese Nuclear Fuel Development (NFD) organization, which has begun testing 304L specimens with neutron irradiation damage levels above 8 dpa.

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: Idaho National Laboratory

Current Partners: Partnerships are being developed for this task

Project Milestones/Deliverables

- Complete initial assessment of key needs for high-fluence IASCC evaluations, *September 2012 – COMPLETED*
- Complete detailed experimental plan, timeline, and assessment of irradiation needs for high-fluence IASCC testing, *February 2013 - COMPLETED*
- Complete revised joint plan with EPRI for very high fluence testing of core internals, *September 2015 - COMPLETED*
- **Complete assessment of the effectiveness of water chemistry changes in mitigating crack growth rate for high fluence samples, *September 2019.***
- Complete report on testing progress on high-fluence effects of IASCC, *on annual basis*

Value of Key Milestones to Stakeholders: Completing a detailed experimental plan for high-fluence IASCC testing (2013) was an essential first step in estimating the impact of IASCC at high fluence. This plan is also critical for building support and partnerships with industry and regulators.

3.3.5 High Fluence Phase Transformations in RPV and Core Internal Materials

The neutron irradiation field can produce large property and dimensional changes in materials. This occurs primarily via one of five radiation damage processes: radiation-induced hardening and embrittlement, phase instabilities from RIS and precipitation, irradiation creep due to unbalanced absorption of interstitials versus vacancies at dislocations, volumetric swelling from cavity formation, and high temperature helium embrittlement due to formation of helium-filled cavities on grain boundaries. For LWR systems, high temperature embrittlement and creep are not common problems due to the lower reactor temperature. However, radiation embrittlement, phase transformation, segregation, and swelling have all been observed in reactor components.

Under irradiation, the large concentrations of radiation-induced defects will diffuse to defect sinks such as grain boundaries and free surfaces. These concentrations are in far excess of thermal-equilibrium values and can lead to coupled-diffusion with particular atoms. In engineering metals such as stainless steel, this results in RIS of elements within the steel. For example, in 316 stainless steel, chromium (important for corrosion resistance) can be depleted at areas, whereas elements like nickel and silicon are enriched to levels well above the starting, homogenous composition.

While RIS does not directly cause component failure, it can influence corrosion behavior in a water environment. Further, this form of degradation can accelerate the thermally driven phase transformations mentioned above and also result in phase transformations that are not favorable under thermal aging (such as gamma or gamma-prime phases observed in stainless steels). Additional fluence may exacerbate radiation-induced phase transformations and should be considered. The wealth of data generated for fast breeder reactor studies and more recently in LWR-related analysis will be beneficial in this effort. However, it is especially important to examine the microstructural differences between experimental fast reactor irradiations and those of lower flux LWR conditions (See Figure 8). This has potential impact on materials properties. Initial data from computational studies coupling thermodynamic and radiation-induced damage models has demonstrated that differences in irradiation flux rate can produce differences in the phase development and stability. New data from ex-service material characterization will be used to validate these models.

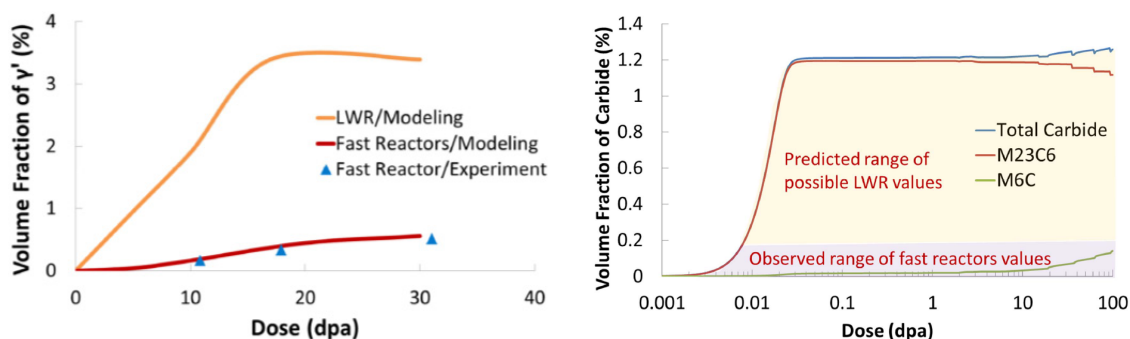


Figure 8: Comparison of the gamma-prime (left) development and carbide formation (right) between fast experimental reactor data and that predicted in lower fluence LWR conditions [Error! Reference source not found.].

Starting in 2015, this task area is also been supportive of the work in understanding precipitation and its effect on properties in high fluence RPV steels. The Cu-rich and Mn-Ni-Si precipitation is a leading cause of embrittlement of RPV steels. To understand this phenomenon, multi-scale modeling approaches are being taken. This previous years work has included developing models for the growth of Mn-Ni-Si precipitates through cluster dynamics methods to assess fraction of precipitate formation from pre-existing Cu-rich precipitate clusters, shown in Figure 9. A separate model for Cu-rich precipitate formation is now being incorporated into the Mn-Ni-Si model for a comprehensive evaluation of the phase development in high fluence RPV steel. A near term plan is to develop of two scales of models for RPV precipitation from the atomic scale and meso scale. Additional effort will focus on the application of cluster dynamics and machine learning methods for predicting hardening behavior as a function of alloy composition and radiation environment (flux, fluence, and temperature). Integrating known hardening mechanisms and mechanical properties models into the cluster dynamics precipitate model to predict precipitation and hardening.

The development and delivery of a validated model for phase transformations in core internal components and reactor pressure vessel steel at high-fluence is an important step in estimating the useful life of components. Understanding which components are susceptible to precipitation of phases detrimental to mechanicals properties is of value to industry and regulators, as it will permit more focused component inspections, component replacements, and more detailed regulatory guidelines.

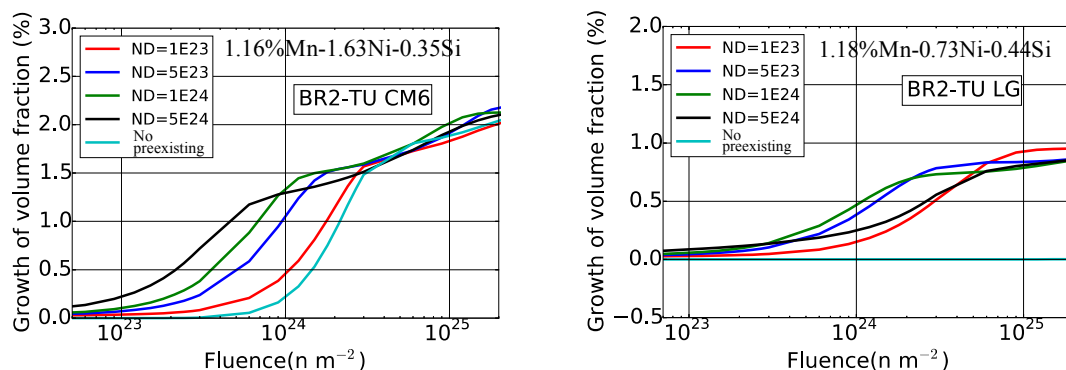


Figure 9: The predicted growth of the Mn-Ni-Si precipitation in irradiated RPV steels of different alloy concentrations assuming different levels of pre-existing Cu-rich precipitation levels, from [Error! Reference source not found.].

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: ORNL with support from University of Wisconsin

Current Partners: EPRI (technical input)

Project Milestones/Deliverables

- Complete report on testing and modeling progress for high-fluence phase transformations, *on annual basis*

- Complete report detailing possible extent of irradiation-induced phase transformations and components of concern, *June 2011 – COMPLETED*
- Complete report detailing initial experimental plan for testing irradiation-induced phase transformations, *August 2011 – COMPLETED*
- Initiate modeling and simulation efforts for prediction of phase transformations in LWR components, *June 2012 – COMPLETED*
- Complete basic model development for phase transformations in LWR components, *April 2014 - COMPLETED*
- Acquire plant-relevant materials for evaluation of high-fluence phase transformations for model benchmarking, *April 2017*
- **Complete modeling of RPV steel hardening as a function of radiation flux, fluence, temperature and alloy composition, *September 2017***
- **Deliver computational tool to model combined thermal and radiation induced segregation of impurity solute elements to grain boundaries in austenitic stainless steels, *March 2018***
- Deliver experimentally validated, physically based thermodynamic and kinetic model of precipitate phase stability and formation in Alloy 316 under anticipated extended lifetime operation of LWRs, *August 2018*
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: The development and delivery for a validated model for phase transformations in core internal components at high-fluence is an important step in estimating the useful life of core internal components. Understanding which components are susceptible to this form of degradation is of value to industry and regulators, as it will permit more focused component inspections, component replacements, and more detailed regulatory guidelines.

3.3.6 High Fluence Swelling of Core Internal Materials

In addition to irradiation hardening processes and diffusion-induced phase transformations, the diffusion of radiation-induced defects can also result in the clustering of vacancies, creating voids that may be stabilized by gas atoms in the material. While swelling is typically a greater concern for fast reactor applications where it can be life-limiting, voids have recently been observed in LWR components such as baffle bolts. The motion of vacancies can also greatly accelerate creep rates, resulting in stress relaxation and deformation. Irradiation-induced swelling and creep effects can be synergistic, and their combined influence must be considered. Longer reactor component lifetimes may increase the need for a more thorough evaluation of swelling as a limiting factor in LWR operation. As above, data, theory, and simulations generated for fast reactor and fusion applications can be used to help identify potentially problematic components.

Irradiation-induced swelling may be severe in core internal components at extended operation. Dimensional changes of core internal components due to irradiation-induced swelling may be life limiting. Longer reactor component lifetimes may increase the need for a more thorough evaluation of

swelling as a limiting factor in LWR operation. This task will provide detailed microstructural analysis of swelling in key samples and components (both model alloys and service materials), including transmission electron microscopy and volumetric measurements. These results will be used to develop and validate a phenomenological model of swelling under LWR conditions. This will be accomplished by extension of past models developed for fast reactor conditions. The data generated and mechanistic studies will be used identify key operational limits (if any) to minimize swelling concerns, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, if necessary, qualify swelling-resistant materials for LWR service.

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: ORNL

Current Partners: EPRI (technical input) and Areva (technical input)

Project Milestones/Deliverables

- Complete report on testing and modeling progress for high-fluence swelling, *on annual basis*
- Complete report detailing possible extent of swelling and components of concern, *June 2011 – COMPLETED*
- Complete report detailing initial experimental plan for testing swelling in LWR components, *August 2011 – COMPLETED*
- Initiate modeling and simulation efforts for prediction of swelling in LWR components, *June 2012 – COMPLETED*
- Complete model development for swelling in LWR components, *December 2014 - COMPLETED*
- Acquire plant-relevant materials for evaluation of swelling for model benchmarking, *May 2017*
- Complete post irradiation testing and examinations of swelling in LWR components and materials, *July 2017*
- **Deliver validated model for swelling in LWR components, August 2017**
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: The development and delivery for a validated model for swelling in core internal components at high fluence (in 2014 and 2017, respectively) is an important step in estimating the useful life of core internal components. Understanding which components are susceptible to this form of degradation is of value to industry and regulators, as it will permit more focused component inspections, component replacements, and more detailed regulatory guidelines.

3.3.7 Cracking Initiation in Ni-base Alloys

Stress corrosion cracking of Ni-base stainless alloys, such as alloy 600 and its weld metals, began to significantly impact PWR performance in the 1980s and led to the need to replace or retire entire steam generators. In addition to primary-side and secondary-side steam generator tubing problems, service cracking of alloy 600 materials has now been documented in many other PWR components, including pressurizer heater sleeves and welds, pressurizer instrument nozzles, reactor vessel closure head nozzles and welds, reactor vessel outlet nozzle welds, and reactor vessel head instrumentation nozzle and welds. Pressurizer nozzles operating at the highest temperature were the first thick-section alloy 600 component identified to crack in service and were typically replaced with austenitic stainless steels. More serious concerns developed when through-wall SCC was found in control rod drive mechanism (CRDM) nozzles in the upper head of the PWR pressure vessels. These extensive problems have resulted in a systematic replacement of the lower Cr, alloy 600 with higher Cr, alloy 690 materials. Although service performance has been excellent for the replacement materials, SCC susceptibility has been identified in the laboratory, prompting continuing questions for long-term component reliability. Stress-corrosion cracking is found in several different forms and may be the limiting factor for extended service. The integrity of these components is critical for reliable power generation in extended lifetimes, and as a result, understanding and mitigating these forms of degradation is very important. Adding additional service life to these components will allow more time for corrosion to occur. The various forms of corrosion must be evaluated as in NUREG/CR-6923 [2] with special attention to those that may be life limiting in extended service.

Cracking in primary piping and steam generator tubing is currently a reliability and safety issue in LWRs and is expected to worsen with additional lifetime. Many forms of cracking are experienced by the Ni-base alloys used as tubing in heat exchangers for nuclear reactor applications. A key outcome of this task is the identification of underlying mechanisms of SCC in Ni-base alloys. Understanding and modeling the mechanisms of crack initiation is a key step in predicting and mitigating SCC in the primary and secondary water circuits. An examination into the influence of surface and metallurgical conditions on precursor states and crack initiation also is a key need for Ni-base alloys (see Figure 10 for an example of the detailed analysis being performed) and austenitic stainless steels. This effort focuses on SCC crack-initiation testing on Ni-base alloys and stainless steels in simulated LWR water chemistries, but includes direct linkages to SCC crack-growth behavior. Carefully controlled microstructure and surface states will be used to generate single-variable experiments. The experimental effort in this task will be highly complementary to efforts being initiated at the Materials Ageing Institute, which are focused primarily on modeling of crack initiation. This mechanistic information could provide key operational variables to mitigate or control SCC in these materials, optimize inspection and maintenance schedules to the most susceptible materials/locations, and potentially define SCC-resistant materials.

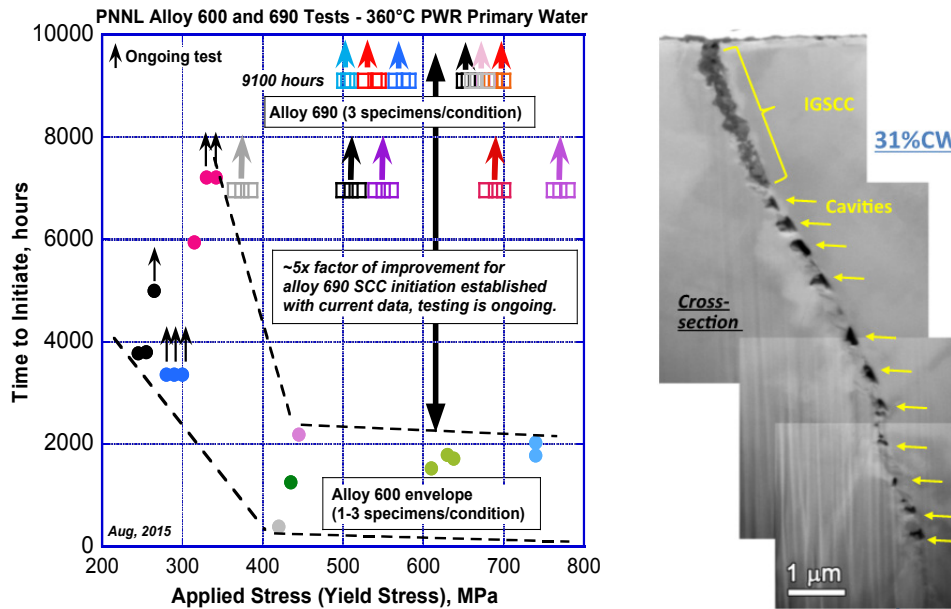


Figure 10: Comparison of the SCC initiation behavior (left) between cold worked alloy 600 and 690. The dashed lines bound the envelope for SCC initiation. Though monitoring of the SCC initiation testing for 690 has not identified crack growth, intergranular (IG) corrosion and cavity formation along grain boundaries (right) is observed as the precursor to crack growth [11,12].

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: PNNL

Current Partners: EPRI (technical input), NRC (technical input and complementary test matrix), Rolls Royce (materials and complementary test matrix)

Project Milestones/Deliverables

- Complete report on testing progress on crack initiation in Ni-base alloys, *on annual basis*
- Complete detailed characterization on precursor states for crack initiation in Ni-base alloys, *March 2012 – COMPLETED*
- Complete Phase 1 mechanistic testing for SCC research, *September 2015 - COMPLETED*
- Initiate predictive modeling and theoretical studies to develop predictive capability for crack initiation in Ni-base alloy piping, *March 2015*
- Complete Phase 2 mechanistic testing for SCC research, *September 2016*
- **Deliver predictive model capability for Ni-base alloy SCC susceptibility, *September 2019***
- Detailed testing and specific subtasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing research to identify the mechanisms and precursor states (2015) is an essential step to predicting the extent of this form of degradation under extended service conditions. Understanding underlying causes for crack initiation may allow for more focused material inspections and maintenance, new SCC-resistant alloys, and development of new mitigation strategies, all of which are of high interest to the nuclear industry. This mechanistic understanding may also drive more informed regulatory guidelines and aging-management programs. In the long term, mechanistic understanding also enables development of a predictive model (2019), which has been sought by industry and regulators for many years.

3.3.8 Environmentally Assisted Fatigue

Fatigue (caused by mechanical or environmental factors, or both) is the number one cause of failure in metallic components. Examples of past experience with this form of degradation in reactor coolant system (RCS) include cracking at the BWR feedwater nozzle; BWR steam dryer support bracket; BWR recirculation pipe welds; PWR surge line to hot leg weld; PWR pressurizer relief valve nozzle welds; PWR cold leg drainline; PWR surge, relief, and safety nozzle-to-safe-end dissimilar metal butt welds; PWR decay heat removal drop line weld; and PWR weld joins at decay heat removal system drop line to a reactor coolant system hot leg. The effects of environment on the fatigue resistance of materials used in operating PWR and BWR plants are uncertain. There is a need to assess the current state of knowledge in environmentally assisted fatigue of materials in LWRs under extended service conditions. It is also important to develop a mechanistic understanding of the role of water chemistry on the microstructural changes in the materials and on their fatigue properties. The objective of this task is to develop a model of environmentally assisted fatigue mechanisms to predict life for this mechanism. This has included analysis of fatigue cycles due to thermal and pressure effects on a system-level analysis. A final report will be delivered in the 2017 to 2021 timeframe, providing both a model of fatigue mechanisms and the supporting experimental data.

In recent work [13], shown in Figure 11, the thermal-mechanical stress analysis of the reactor pressure vessel finite element (FE) model was investigated based on experimentally determined material properties from 2015. Stress profiles associated with the nozzle area of the RPV was examined over time during heat up conditions to full power operation under simulated load following conditions. Analysis was conducted with and without the presence of preexisting cracks in the reactor nozzles (both axial as shown in Figure 11, and circumferential cracks were investigated) under a simulated load following condition. While the preexisting crack did not grow under the load conditions, its presence created larger stress distributions that could further influence primary water stress corrosion cracking. Development of these models continues.

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: ANL

Current Partners: EPRI (technical input via Environmentally Assisted Fatigue Working Group)

Project Milestones/Deliverables

- Complete report on testing progress on environmentally assisted fatigue, *on annual basis*
- Initiate modeling and simulation efforts for prediction of environmentally assisted fatigue in LWR components, *January 2012 – COMPLETED*
- Complete base model development for environmentally assisted fatigue in LWR components, *August 2015 -COMPLETED*
- **Complete experimental validation and deliver model for environmentally assisted fatigue in LWR components, August 2017**
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

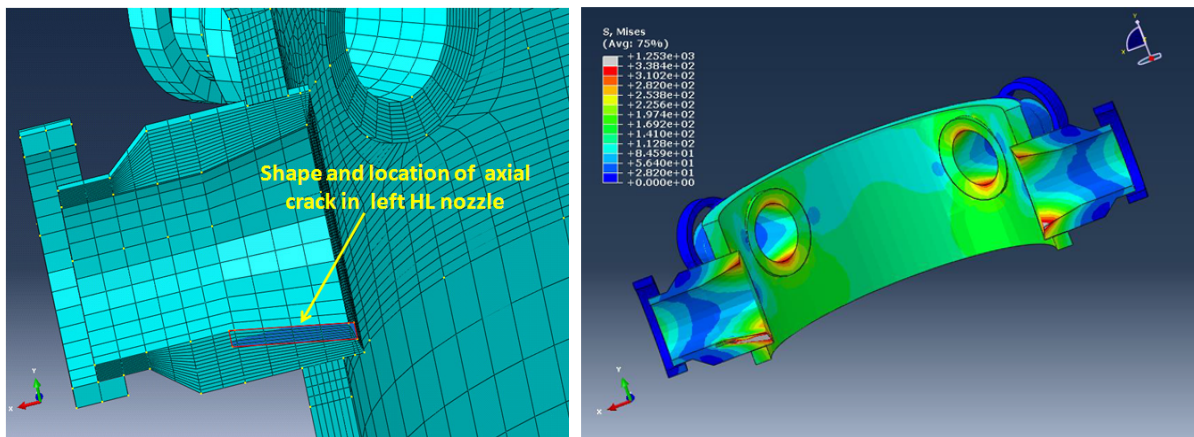


Figure 11: Shape and location of axial crack in left HL nozzle of RPV (left) , with Von-Mises stresses under typical full power condition [13] (right).

Value of Key Milestones to Stakeholders: Completing research to identify the mechanisms of environmentally assisted fatigue to support model development (2014) is an essential step to predicting the extent of this form of degradation under extended service conditions. This issue has been identified as a key need by regulators and industry. Delivering a model for environmentally assisted fatigue (2017) will enable more focused material inspections, material replacements, and more detailed regulatory guidelines.

3.3.9 Thermal Aging of Cast Stainless Steels

The cast austenitic stainless steels (CASSs) are highly corrosion-resistant iron-chromium-nickel alloys with a duplex austenite and ferrite structure and have been used for a variety of applications in nuclear power plants. The CASSs are important materials in modern LWR facilities since a massive amount of the alloy is used for the majority of the pressure-boundary components in reactor coolant systems. Relatively few critical degradation modes of concerns are expected within the current designed lifetime of 40 years given that the CASS components have been processed properly. Today's fleet has experienced very limited failures or material degradation concerns. In the limited number of service observations of degradation, all have been attributed to some abnormal characteristics due to high carbon content or improper processing.

Under extended service scenarios, there may be degradation modes to consider for the CASSs and components at temperatures much closer to operation temperatures. A prolonged thermal aging could lead to decomposition of key phases and formation of other deleterious phases. Such aging could result in the loss of fracture toughness (analogous to that observed in other martensitic stainless steels). The properties of CASS are strongly dependent on the amount of ferrite, which may vary based on composition and processing conditions. Additional surveys of potential phase changes and aging effects would help reduce uncertainty of these mechanisms.

In this research task, the effects of elevated temperature service in CASS will be examined. The possible effects of phase transformations that can adversely impact mechanical properties will be explored. Preliminary data from 1,500 hour aging of model CASS alloys (CF3, CF3M, CF8 and CF8M) shows a strong effect of aging on the transition temperature at which fracture energy is reduced to 41 Joules (T_{41J}), Figure 12. However, the T_{41J} transition temperature still remains far below room temperature.

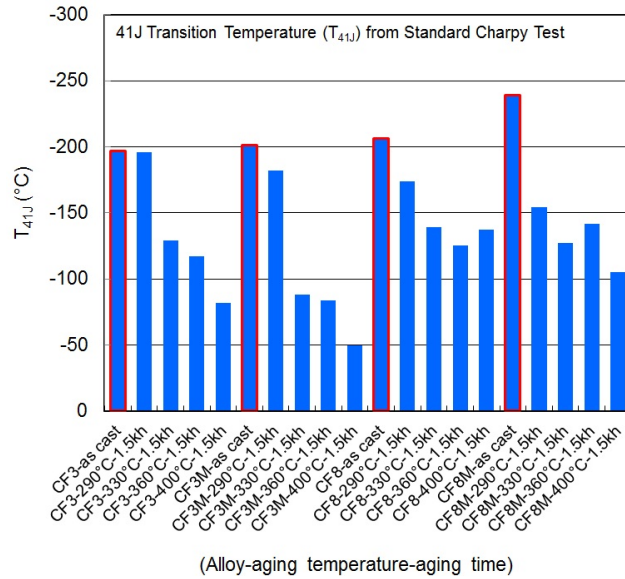


Figure 12: The effect of 1,500 hour aging at 290, 330, 360 and 400 °C on the T_{41J} transition temperature of model CASS alloys of CF3, CF3M, CF8 and CF8M [14].

Mechanical and microstructural data obtained through accelerated aging experiments and computational simulation will be the key input for the prediction of CASS behaviors and for the integrity analyses for various CASS components. While accelerated aging experiments and computational simulations will comprise the main components of the knowledge base for CASS aging, the data will also be obtained from operational experience. This data is required to validate the accelerated aging methodology. In addition to using existing database, therefore, a systematic campaign to obtain mechanical data from used materials or components will be pursued. Further, the detailed studies on aging and embrittlement mechanisms as well as on deformation and fracture mechanisms are performed to understand and predict the aging behavior over extended lifetime. The final output, this task is expected to provide conclusive

predictions for the integrity of the CASS components of LWR power plants during the extended service life beyond 60 years.

Product: High quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: PNNL

Current Partners: EPRI (technical input), Korean Advanced Institute of Science and Technology (through INERI)

Project Milestones/Deliverables

- Complete report on testing progress for cast-stainless steel aging, *on annual basis*
- Complete plan for development of cast stainless steel aging, *September 2012 – COMPLETED*
- Complete report on testing progress for cast stainless steel components, *on annual basis*
- Initiate accelerated aging experiments, *March 2013 - COMPLETED*
- Complete development of computational tools and deliver preliminary ageing simulations for cast stainless steels, *September 2014 – COMPLETED*
- Complete 10,000 hour aging of CASS model alloys, EPRI provided archival materials and wrought comparison alloys, *June 2016 - COMPLETED*
- **Complete analysis and simulations on aging of cast stainless steel components and deliver predictive capability for cast stainless steel components under extended service conditions, *September 2018***

Value of Key Milestones to Stakeholders: Completing research to identify potential thermal aging issues for cast stainless steel components (2017 and 2018) is an essential step to identifying possibly synergistic effects of thermal aging (corrosion, mechanical, etc.) and predicting the extent of this form of degradation under extended service conditions. Understanding the mechanisms of thermal aging will enable more focused material inspections, material replacements, and more detailed regulatory guidelines. This data will also help close gaps identified in the EPRI MDM and upcoming EMDA reports.

3.4 Concrete

As concrete ages, changes in its properties will occur as a result of continuing microstructural changes (e.g., slow hydration, crystallization of amorphous constituents, and reactions between cement paste and aggregates), as well as environmental influences. These changes must not be so detrimental that the concrete is unable to meet its functional and performance requirements. Concrete, however, can suffer undesirable changes with time because of improper specifications, a violation of specifications, adverse performance of its cement paste matrix, or adverse environmental influence on aggregate constituents. Changes to the embedded steel reinforcement as well as its interaction with concrete can also be detrimental to concrete's service life.

Figure 13 serves as a reminder that large areas of most reactors have been constructed by use of concrete. In general, the performance of reinforced concrete structures in nuclear power plants has been very good. Although the vast majority of these structures will continue to meet their functional or performance requirements during the current and any future licensing periods, it is reasonable to assume that there will be isolated examples where, as a result primarily of environmental effects, the structures may not exhibit the desired durability (e.g., water-intake structures and freezing/thawing damage of containments) without some form of intervention.

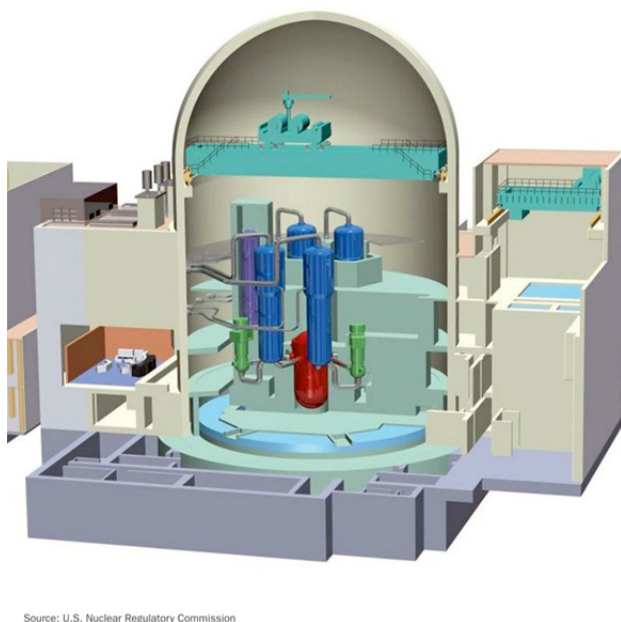


Figure 13: Cut-away of a typical pressurized water reactor, illustrating large volumes of concrete and the key role of concrete performance.

3.4.1 Concrete and Civil Structure Degradation

Although activities by several regulatory authorities have addressed aging of nuclear power plant structures (e.g., Nuclear Regulatory Commission, Nuclear Energy Agency, and International Atomic Energy Agency), additional structure-related research is needed in several areas to demonstrate that the structures will continue to meet functional and performance requirements (e.g., maintain structural margins). Structural research topics include (1) compilation of material property data for long-term performance and trending, evaluation of environmental effects, and assessment and validation of NDE methods; (2) evaluation of long-term effects of elevated temperature and radiation; (3) improved damage models and acceptance criteria for use in assessments of the current as well as the future condition of the structures; (4) improved constitutive models and analytical methods for use in determining nonlinear structural response (e.g., accident conditions); (5) nonintrusive methods for inspection of thick, heavily reinforced concrete structures and basemats; (6) global inspection methods for metallic pressure boundary components (i.e., liners of concrete containments and steel containments) including inaccessible areas and the back side of liner; (7) data on application and performance (e.g., durability) of repair materials and techniques; (8) utilization of structural reliability theory incorporating uncertainties to address time-dependent changes to structures to ensure that minimum accepted performance requirements are exceeded.

and to estimate ongoing component degradation to estimate end-of-life; and (9) application of probabilistic modeling of component performance to provide risk-based criteria to evaluate how aging affects structural capacity.

Activities under the LWRS program presently are being conducted under Tasks 1, 2, 3, 4, and 5. Complementary activities are being conducted under an NRC program at ORNL, addressing Task 2. EPRI has activities under Tasks 2, 3, and 4. Task 7 is being addressed by the Nuclear Energy Standards Coordination Collaborative headed by the National Institute for Standards and Technology.

In the past three years, irradiation effects in concrete have been the focus of considerable thought and research. Overtime, the properties of concrete change due to ongoing changes in the microstructure driven by radiation conditions (spectra, flux, fluence), temperature, moisture content in the material and loading conditions. These changes in properties have been considered minimal to the integrity of concrete structures in nuclear power plants during the original operational timeline of 40 years. However, the current understanding of radiation induced degradation mechanisms is insufficient to determine the properties of irradiated concrete structures in LWRs when the reactor life is extended beyond 40 or 60 years. Further, even the levels of irradiation that the concrete structures may experience have significant uncertainties. Work within the program is addressing the radiation induced property changes of both the minerals that are contained within the aggregate structure as well as concrete paste and mixes. Furthermore, information obtained from this testing is being cataloged in the Irradiated Materials Aggregates Concrete (IMAC) database with information on representative microstructures of the concrete mixes being studied, which will include information from plant structures. Work in the past year has also characterized and bound irradiation limits for both neutron and gamma irradiation, and began the development of a unified damage parameter.

Significant accomplishments have also been achieved in the development of irradiation damage models centered on the meso-scale to evaluate the damage initiation and development of cracks following exposure to constant or varying radiation and temperature conditions as well as load. An example of the damage generated in the concrete for a previous irradiation experiment is shown in Figure 14. Here pre-irradiation drying produces very little crack initiation despite contraction of the paste, but exposure to neutron irradiation results in the development of cracking due to radiation induced swelling of the aggregates [15].

Another mode of degradation being evaluated for its impact on structural concrete performance is that of alkali-silica reactions (ASR) that can produce swelling of the concrete paste resulting in cracking and weakening of the shear capacity of the concrete structure. It is the research goal of this project is to study the development of ASR expansion and induced damage of large-scale specimens representative of structural concrete elements found in nuclear power plants. This will be through experimentally validated models that explore the structural capacity of ASR affected structures like the biological shield, the containment building and fuel handling building. Experimental testing will be conducted in accelerated conditions, employing extensive monitoring and nondestructive techniques to evaluate structural stresses generated in the large block test specimens. An example of the testing includes the ASR Test Assembly (Figure 15), which will provide an opportunity to monitor the development of ASR under accelerated conditions in very large representative structures. The development of ASR will be monitored by both

passive and active NDE techniques. Following conclusion of the monitoring program, final destructive testing will be performed to address the question of the shear capacity of the ASR affected concrete.

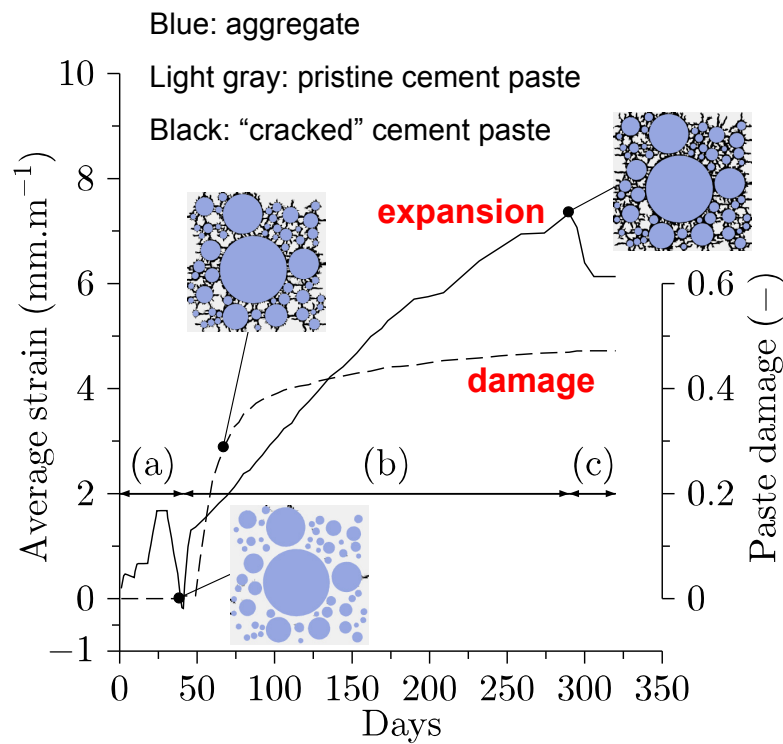


Figure 14: Numerical model accounting for the effects of neutron irradiation on concrete at the mesoscale with simulated experimental test reactor conditions. Work examines the effects of thermal and irradiation aging that accounts for shrinkage (in 'a' during the thermal hold) and volumetric thermal expansion (during 'b' irradiation). Damage resulting from crack development in the paste is due to thermal and radiation-induced effects. Irradiation temperature was 223 °C and total simulated fluence was 1×10^{19} n/cm² (E>1 MeV) [15].

To support these activities, a detailed and populated database on concrete performance, with data for performance into the first life-extension period, high-temperature effects, and irradiation effects, will be delivered by 2016. Plans for research at EPRI and NRC will continue to be evaluated to confirm the complementary and cooperative nature of concrete research under the MAaD pathway. In addition, the formation of an Extended Service Materials Working Group for concrete issues will provide a valuable resource for additional and diverse input.



Figure 15: The ASR Test Assemblies at the University of Tennessee, during concrete pouring (left). Foreground shows the steel frame for the constrained ASR test condition, with the middle mold of the ASR affected block for unconstrained ASR testing being poured, and the background mold of one of the unconstrained non-ASR affected blocks. Small test specimens (right image) were also prepared for periodic mechanical testing during the aging process. The ASR test blocks contain $1\frac{3}{8}$ -inch diameter steel rebar spaced at 10 inches and will be located in an environmental chamber enclosure (to be built following concrete pour) that maintains 100 °F and 95% relative humidity.

Product: Development of a worldwide database on concrete performance, high quality data, and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities, support development of detailed understanding of irradiation effects on concrete and civil structures.

Lead Organization: ORNL

Current Partners: EPRI (technical input, Irradiated Concrete Working Group), NRC (technical input, Irradiated Concrete Working Group), Materials Ageing Institute (MAI) (technical input, Irradiated Concrete Working Group).

Project Milestones/Deliverables

- Complete report on testing progress for concrete performance, *on annual basis*
- Initiate collaborative program with EPRI and MAI on concrete degradation research, *March 2011 – COMPLETED*

- Completion of concrete database framework, *August 2011 – COMPLETED*
- Provide field data and results to MAI for benchmarking of the MAI concrete performance models, *November 2011 – COMPLETED*
- Complete validation of data contained in the concrete performance database and place database in public domain, *December 2013 - COMPLETED*
- Initiate single-variable irradiation campaign to assess radiation-induced volumetric expansion of key aggregate types – *December 2014 - COMPLETED*
- **Deliver unified parameter to assess irradiation-induced damage in concrete structures, *September 2016***
- **Complete initial / limited model to assess the impact of radiation on structural performance for concrete components, *December 2017***
- Complete characterization of radiation damage in concrete materials, *March 2019*
- **Complete model tool to assess the combined effects of radiation and alkali-silica reactions on structural performance for concrete components, *December 2020***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing and publishing a database of concrete performance (2013) will yield a high-value tool accessible to all stakeholders. This will allow for more focused research on remaining knowledge gaps and enable more focused material inspections. These tools are of high value to industry as they are partners in their development.

3.4.2 Nondestructive Evaluation of Concrete and Civil Structures

Developing new nondestructive examination (NDE) techniques that allow for the condition monitoring of concrete structures and components is the objective of this work. Much of the NDE techniques initially available were developed for concrete structures such as bridges and other thinner structures as compared to the biological shield of a nuclear plant. This effort includes performing a survey of available samples, developing techniques to perform volumetric imaging on thick reinforced concrete sections, determining physical and chemical properties as a function of depth, developing techniques to examine interfaces between concrete and other materials, developing acceptance criteria – model and validation, and developing automated scanning systems. An initial step in this R&D plan is to examine the key issues and available technologies. Key issues for consideration can include new or adapted techniques for concrete surveillance. Specific areas of interest include reinforcing steel condition, chemical composition, strength, or stress state. Recent developments have focused on new data processing techniques, such as Model-Based Image Reconstruction (MBIR) forward model development. This nonlinear model is effective when examining non-homogeneous material. An example of the results from consecutive signal processing iterations of adjusted signal parameters are shown in Figure 16, where the object within the concrete begins to become identifiable.

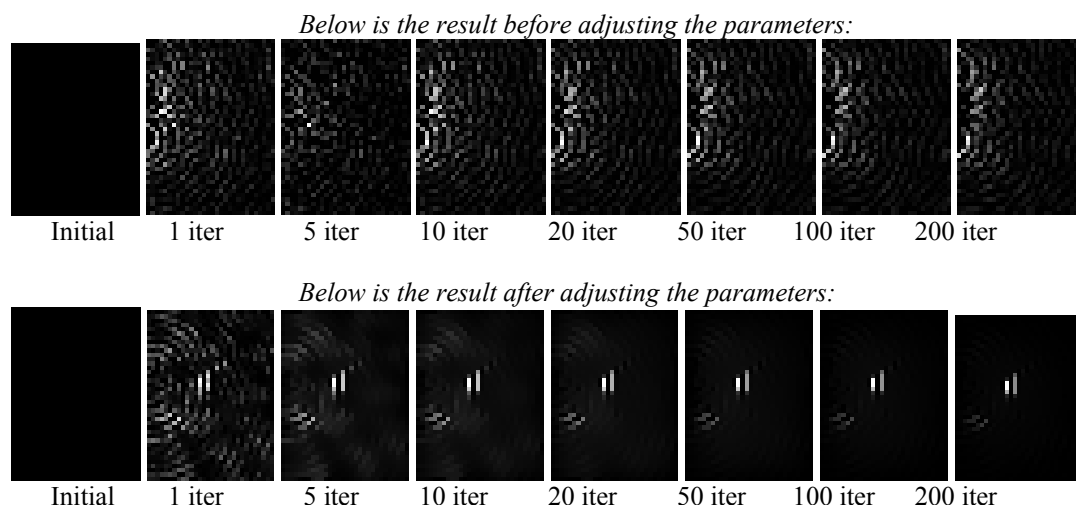


Figure 16: Signal processing of the Model Based Image Reconstruction forward model development using signal processing iterations, from [16].

Product: New monitoring techniques and tools, and complementary data to support mechanistic studies

Lead Organization: ORNL

Current Partners: Partnerships are being developed for this new task.

Project Milestones/Deliverables

- Complete report on testing progress for concrete and civil structures, *on annual basis*
- Complete plan for development of RPV NDE technologies, *September 2012 – COMPLETED*
- Produce first volumetric image of thick concrete sections as part of NDE development, *June 2014 - COMPLETED*
- Produce preliminary model for critical defects in concrete based on NDE results (leveraging current modeling approaches and using data from other LWRS projects), *December 2015*
- **Complete prototype proof-of-concept system for NDE of concrete sections, *September 2017***
- Complete preliminary design of deployment system for concrete NDE, *September 2019*
- **Complete prototype of concrete NDE system, *September 2019***

Value of Key Milestones to Stakeholders: The development of NDE techniques to permit monitoring of the concrete and civil structures could be revolutionary and allow for an assessment of performance that is not currently available via core drilling in operating plants. This would reduce uncertainty in safety margins and is clearly valuable to both industry and regulators.

3.5 Cabling

A variety of environmental stressors in nuclear reactors can influence the aging of low- and medium-electrical-power and instrumentation and control (I&C) cables and their insulation, such as temperature, radiation, moisture/humidity, vibration, chemical spray, mechanical stress, and oxygen present in the surrounding gaseous environment (usually air). Exposure to these stressors over time can lead to degradation that, if not appropriately managed, could cause insulation failure, which could prevent associated components from performing their intended safety function.

Operating experience has demonstrated failures of buried medium-voltage ac and low-voltage dc power cables due to insulation failure. NRC's Generic Letter (GL) 2007-01 indicates that low-voltage cables have failed in underground applications and that the cable failures were due to a variety of causes, including manufacturing defects, damage caused by shipping and installation, exposure to electrical transients, and abnormal environmental conditions during operation. While current causes for cable failures in nuclear plants has been primarily related to mechanical and physical damage, as well as human error [17], aging of reactors is expected to see higher incidents of failure in areas of irradiation, thermal and moisture related stresses.

As a result, cable aging is a concern that currently faces the operators of existing reactors. The plant operators carry out periodic cable inspections using nondestructive examination techniques to measure degradation and determine when replacement is needed. Physical degradation of these cables is primarily caused by long-term exposure to high temperatures. Additionally, stretches of cables that have been buried underground are frequently exposed to groundwater. Wholesale replacement of cables would likely be a "show stopper" for plant operation beyond 60 years because of the cost and difficulty in replacement.

The two primary activities for cable aging research in LWRS are listed below along with key outcomes for each task.

- Mechanisms of cable degradation: provides understanding of role of material type (i.e., EPR, XLP, etc.), history, and environment on cable insulation degradation; understanding of accelerated testing limitations; support to partners in modeling activities, surveillance, and testing criteria
- Techniques for NDE of cables: provides new technologies to monitor material and component performance

The technical approach to evaluating cable lifetime is shown in Figure 17, which utilizes harvested and representative cables that are historically similar cable formulations used in reactors that were stored appropriately and not used in-reactor service. Testing involves the isolation of the effects of various environmental stressors, but also the synergistic effects that create mechanical, physical and electrical property changes due to chemical changes in the insulation. These changes are also being evaluated by NDE techniques to develop methods suitable for in-field condition monitoring. The ultimate goal of the accelerated aging testing and NDE evaluations are to determine remaining cable useful life.

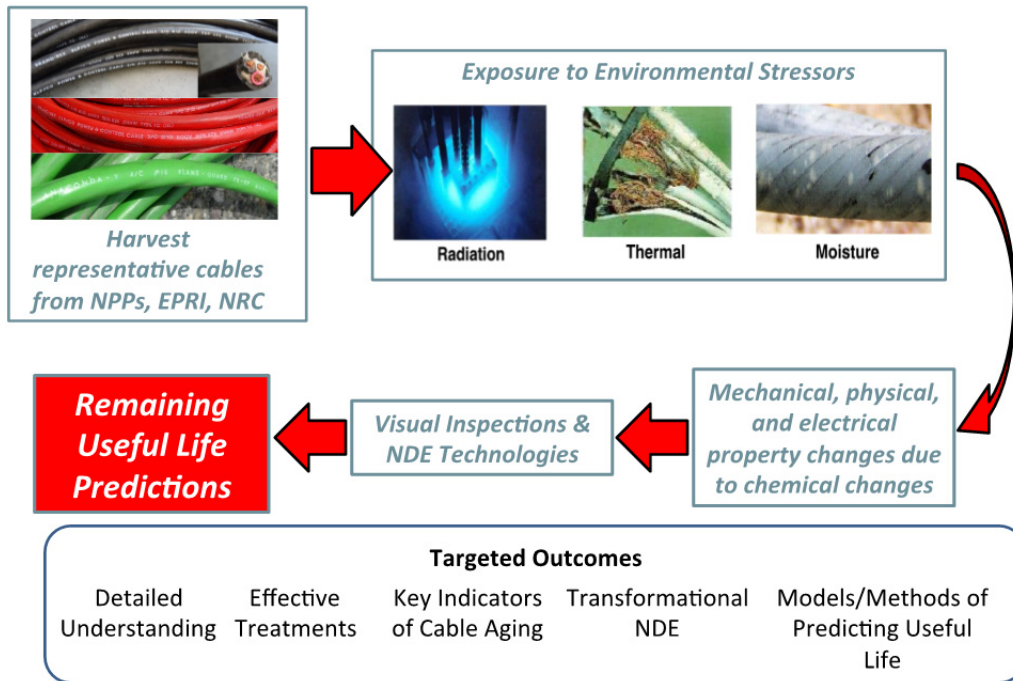


Figure 17: Diagram of the technical approach to cable aging studies to understand the different degradation modes affecting cable lifetime and to evaluate deployable NDE methods for determining remaining useful life [18].

3.5.1 Mechanisms of Cable Insulation Aging and Degradation

The motivation for R&D in this area comes from the need to address the aging management of in-containment cables at nuclear reactors. With nearly 1000 km of power, control, instrumentation, and other cable types typically found in a nuclear reactor, it would be a significant undertaking to inspect all of the cables. Degradation of the cable jacket, electrical insulation, and other cable components are key issues for assessing the ability of the currently installed cables to operate safely and reliably for another 20 to 40 years beyond the initial operating life.

Currently, little or no data exists on long-term cable performance in nuclear power plants. To ensure reliable operation of sensors, controls, and monitoring systems, cable lifetimes and degradation must be understood. This task will begin to estimate expected lifetime of medium- and low-voltage cabling that operates in a wetted environment, using laboratory testing and in-service components for evaluation. The first evaluation will provide a critical assessment of testing needs and proper roles for DOE-led research.

Mechanisms of cable degradation provide an understanding of the role of material type, history, and the environment on cable insulation degradation; understanding of accelerated testing limitations; and support to partners in modeling activities, surveillance, and testing criteria. This task will provide experimental characterization of key forms of cable and cable insulation in a cooperative effort with NRC and EPRI. Tests will include evaluations of cable integrity following exposure to elevated temperature, humidity, and/or ionizing irradiation. This experimental data will be used to evaluate mechanisms of cable aging and determine the validity or limitations of accelerated aging protocols. The experimental data and

mechanistic studies can be used to help identify key operational variables related to cable aging, optimize inspection and maintenance schedules to the most susceptible materials/locations, and, in the long-range, design tolerant materials.

Recent work has concentrated on addressing the gaps in the knowledge of cable degradation and include the synergistic thermal and irradiation effects on cable degradation, inverse temperature effects, accurate determination of activation energies for specific forms of degradation, dose rate effects, diffusion limited oxidation and the development of NDE techniques. Accomplishments include the completion of high dose rate thermal and irradiation testing of cross-linked polyethylene and chlorosulfonated polyethylene. This was accomplished through using facilities at both PNNL and ORNL on cables procured by EPRI through the nuclear power utilities and consists of both “new old stock” (old cables stored on site, but not used in service) and cables removed from service. The current standard measurement of degradation is normally tested through tensile tests of aged polymer. An accepted point of reference for significant materials degradation has been established to be 50% ultimate tensile elongation. For the case of control rod cable that has seen in-service, Figure 18, thermal aging reveals a trend in the elongation at break below that of non-irradiated cable.

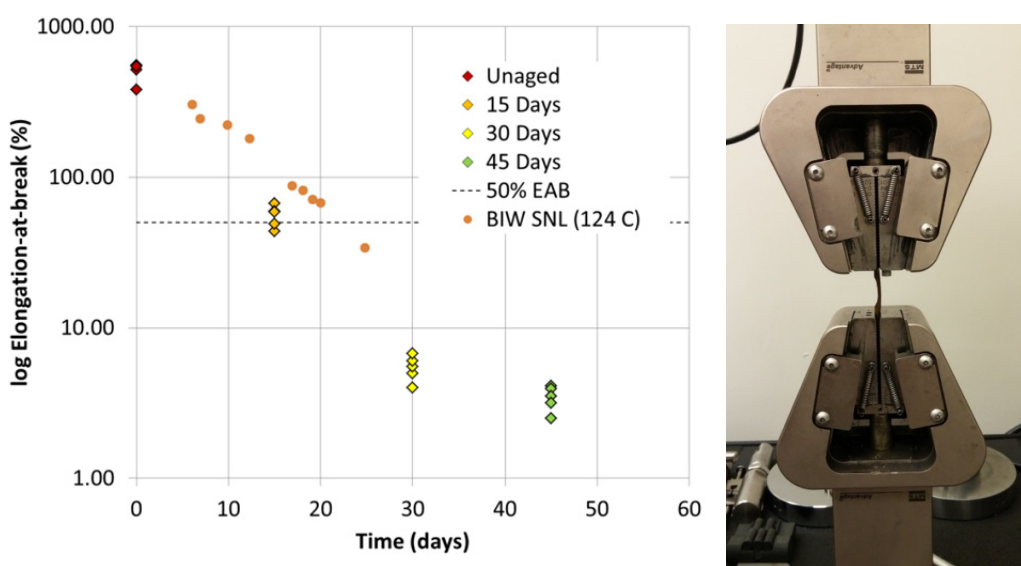


Figure 18: Trend in elongation at break for 124 °C thermally aged hypalon jacked ethylene-propylene rubber insulated cable comparing pre-irradiated service material to that of non-irradiated.

Product: Assessment of accelerated testing techniques; high quality data and mechanistic understanding delivered via reports and technical papers; support for model and simulation activities

Lead Organization: PNNL with support from ORNL

Current Partners: EPRI (technical input and complementary research), NRC (technical input and complementary research)

Project Milestones/Deliverables

- Complete report on testing progress for cable insulation aging and degradation, *on annual basis*
- Complete report detailing highest priority needs and concerns for future testing of cable insulation, *September 2010 – COMPLETED*
- Initiate testing on key degradation issues for cabling and cable insulation, *November 2010 – COMPLETED*
- Initiate evaluation of possible mitigation techniques for cable insulation degradation, *March 2011 – COMPLETED*
- Acquire relevant plant cable insulation for additional testing, *June 2012 – COMPLETED*
- **Complete key analysis of key degradation modes of cable insulation, August 2017**
- **Complete assessment of cable mitigation strategies, December 2018**
- Begin benchmarking of cable degradation model, *March 2018*
- **Deliver predictive model for cable degradation, August 2019**
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Completing research to identify and understand the modes of cable insulation degradation are an essential step to predicting the performance of cable insulation under extended service conditions. This data is clearly critical to developing and delivering a predictive model for cable insulation degradation (2019). Both will enable more focused inspections, material replacements, and more informed regulations. The development of in-situ mitigation strategies may also allow for an alternative to cable replacement and would be of high value to industry by avoiding costly replacements.

3.5.2 Nondestructive Evaluation of Cable Insulation

The most important criteria for cable performance is its ability to withstand a design basis accident. With nearly 1000 km of power, control, instrumentation, and other cable types typically found in an NPP, it would be a significant undertaking to inspect all the cables. Degradation of the cable jacket, electrical insulation, and other cable components is a key issue that is likely to affect the ability of the currently installed cables to operate safely and reliably for another 20 to 40 years beyond the initial operating life. The development of one or more NDE techniques and models that could assist in determining the remaining life expectancy of cables or their current degradation state would be of significant interest. The ability to nondestructively determine material and electrical properties of cable jackets and insulation without disturbing the cables or connections is essential.

The objectives of this task include the development and validation of new NDE technologies for the monitoring of condition of cable insulation. This task area delivered an R&D plan in 2012 for sensor development to monitor reactor metal performance. An initial step in this R&D plan is to examine the key issues and available technologies. In future years, this research will include an assessment of key aging indicators, development of new and transformational NDE methods for cable insulation, and development of a method for utilizing the NDE signals and mechanistic knowledge from other areas of the LWRS

program to provide predictions of remaining useful life. A key element underpinning these three thrusts will be the harvesting of aged materials for validation.

Product: New monitoring techniques and tools, and complementary data to support mechanistic studies

Lead Organization: PNNL

Current Partners: Partnerships are being developed for this new task.

Project Milestones/Deliverables

- Complete report on testing progress for cable insulation NDE, *on annual basis*
- Complete plan for development of cable insulation NDE technologies, *September 2012 – COMPLETED*
- **Complete assessment of cable insulation precursors to correlate with performance and NDE signals, *September 2016***
- **Demonstrate field testing of prototype system for NDE of cable insulation, *September 2017***
- **Deliver predictive capability for end of useful life for cable insulation, *September 2019***

Value of Key Milestones to Stakeholders: The development of NDE techniques to permit in-situ monitoring of the cable insulation performance could be revolutionary and allow for an assessment of cable insulation performance at specific locations of interest and at more frequent intervals, a significant difference from today's methodology. This would reduce uncertainty in safety margins and is clearly valuable to both industry and regulators.

3.6 Buried Piping

Maintaining the many miles of buried piping at a reactor is an area of concern when evaluating the feasibility of extended plant operations. While much of the buried piping is associated with either the secondary side of the plant or other non-safety-related cooling systems, some buried piping serves a direct safety function. Maintaining the integrity of the buried piping in these systems is necessary to ensure the systems can continue to perform their intended functions under extended plant service periods. Industry and regulators already are performing considerable work in this area. The LWRS program continues to evaluate this area for gaps and needs relative to extended service.

3.7 Mitigation Technologies

Mitigation technologies include weld repair, post-irradiation annealing, water chemistry modifications to reduce SCC, but may also include the utilization of new materials that provide superior resistance to the harsh LWR conditions. Welding is widely used for component repair. Weld-repair techniques must be able to be utilized for irradiated materials that contain levels of helium from transmutation reactions.

Furthermore, welding techniques need to be resistant to continued degradation under LWR conditions. One of the research areas under Mitigation Technologies is the development of new welding techniques, weld analysis, and weld repair. A critical assessment of the most advanced methods and their viability for LWR repair weld applications is needed. Additional work on mitigating hardening effects in RPV materials through post-irradiation annealing may be a means of extending operational life. Water chemistry modification is another mitigation technology that warrants evaluation, and is currently being pursued in both the IASCC related tasks within the LWRS program, as covered earlier. The other mitigation strategy is to evaluate the advanced austenitic, ferritic-martensitic and oxide dispersion strengthened steels, as well as other Ni-base alloys as potential replacement alloys for more commonly used materials. Some of the materials of interest have seen attention for other advanced fission and fusion reactor concepts, providing a starting base for irradiated materials data from which to draw upon.

The primary activities in LWRS-supported mitigation technologies are listed below along with key outcomes for each task.

- Weld repair: provides understanding and model of helium effects under welding, validation of residual stress models currently under development, and deployment of advanced repair welding techniques
- Thermal annealing: provides critical assessment of thermal annealing as a mitigation technology for RPV and core internal embrittlement and research to support deployment of thermal annealing technology
- Advanced replacement alloys: provides new alloys for use in LWR application that provide greater margins and performance, and support to industry partners in their programs

Each task is described in more detail in the sections that follow.

3.7.1 Advanced Weld Repair

Welding is extensively used in construction of nuclear reactor components and subsystems. The performance of weldments (including both weld metal and the adjacent heat affected zone) is critical to the safe and efficient operation of the nuclear reactor. Weldments frequently are the most susceptible locations for corrosion, stress-corrosion, and mechanical failures. Weld repairs are a potential method for mitigating cracking or degradation instead of component replacement. With extended lifetimes and increased repair frequency, these welds must be resistant to corrosion, irradiation, and other forms of degradation. EPRI's recent strategic plan for long-term nuclear reactor operation identifies two critical long-standing welding-related technical challenges requiring further R&D.

Today, welding is widely used for repair, maintenance, and upgrade of LWR components. These repair welds need to have improved resistance to SCC and to other long-term degradation. New and improved welding techniques (processes and techniques) are needed to avoid and/or reduce any deleterious effects associated with the traditional welding fabrication practices. Advances in welding technology have been significant in the past two decades, both in process technology and knowledge of welding residual stress control, and some are candidates for further development. Specifically, the following areas should be

evaluated: (1) proactive weld residual stress control and mitigation techniques through welding process innovation and/or post-weld treatment; (2) welding technology to repair irradiated reactor internals to avoid helium-induced cracking during welding repair; (3) improved weld metal development; and (4) new solid-state joining processes, such as friction stir welding, and high-energy welding, such as laser welding for microstructure and residual stress benefits. Development of new and improved welding technology for weld residual stress and microstructure control will require better understanding and predictive capability.

The objective of this task is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without helium induced cracking. Research includes mechanistic understanding of helium effects in weldments. This modeling task is supported by characterization of model alloys before and after irradiation and welding. This model can be used by stakeholders to further improve best practices for repair welding for both existing technology and advanced technology. In addition, this task will provide validation of residual stress models under development using advanced characterization techniques such as neutron scattering. Residual stress models also will improve best practices for weldments of reactors today and under extended service conditions. These tools could be expanded to include other industry practices such as peening. Finally, advanced welding techniques (such as friction-stir welding, laser welding, and hybrid techniques) will be developed and demonstrated on relevant materials (model and service alloys). Characterization of the weldments and qualification testing will be an essential step. To realize this step, a unique facility has been constructed in partnership with EPRI. A welding station was engineered and recently installed to support friction stir welding and laser welding development on irradiated materials. The welding cubicle is located at the Radiochemical Engineering Development Center (REDC) hot cell facility at ORNL. Images of the installation of this cubicle are shown in Figure 19, which will be ready for testing on irradiated materials in FY17.

The objective of this task is to develop advanced welding technologies that can be used to repair highly irradiated reactor internals without helium induced cracking. Toward this goal, a new in-situ stress management approach for controlling temperature and strain distribution round weld pool was developed. The in-situ temperature and strain distribution were measured by digital image correlation (DIC) and infrared (IR) thermography respectively [19]. In addition, a computational model that can be used to gain a fundamental understanding of the effect of welding stress and temperature on the formation helium induced cracking during welding, and the effect of the auxiliary heating on stress and temperature distribution was developed. These are shown in Figure 20 below. It is noted that the technology developed in this task is under patent application. As such, specific details in the technology are omitted. Nevertheless, the effectiveness of the proactive in-situ stress management technology is illustrated below.

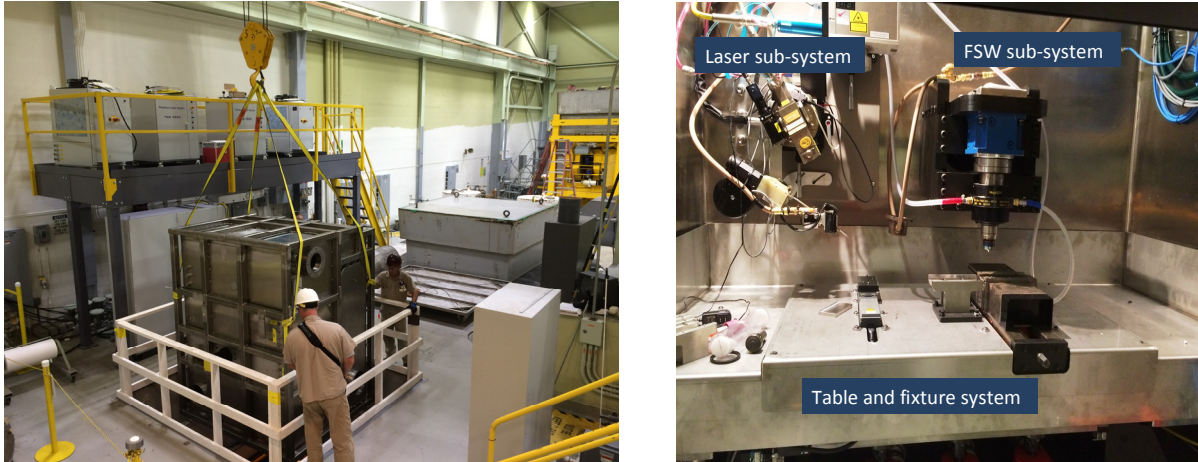


Figure 19: The positioning of the irradiated materials welding cubicle through the roof access port of a cell bay at the REDC facility (laser welding power supply and control units are shown in the mezzanine structure in the background), and view of the laser and friction stir welding subsystems inside the cubicle.

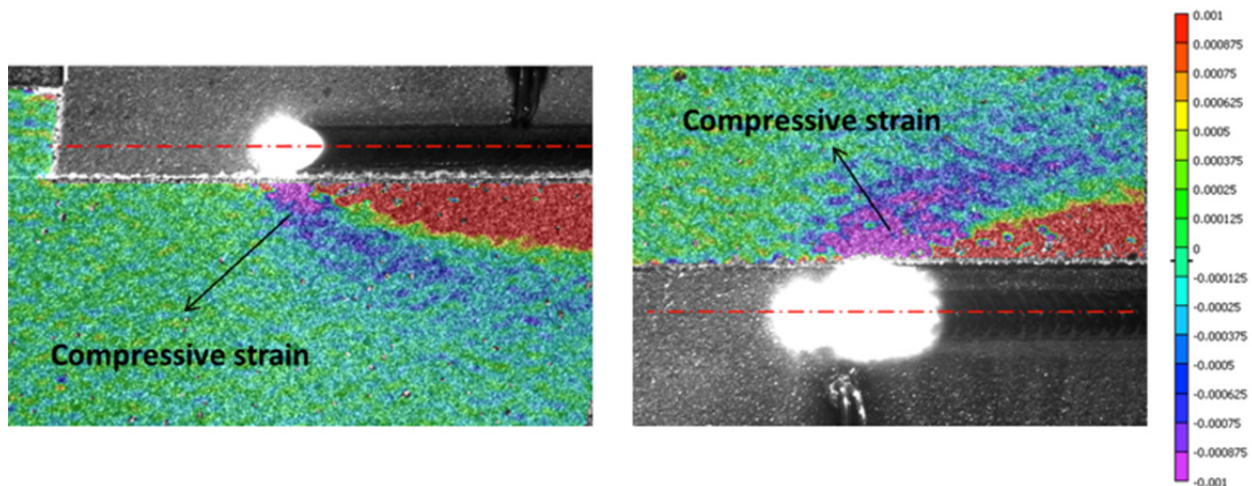


Figure 20: Total transverse strain using advanced residual stress management (welding speed at 15mm/s): without (left) and with (right) stress management approach. The area of compressive strain is clearly increased with this approach [19].

Product: Development of new welding techniques, high quality data on weld performance, mechanistic understanding of welding of irradiated materials, and model capability for residual stress management

Lead Organization: ORNL

Current Partners: EPRI (cost-sharing and technical input)

Project Milestones/Deliverables

- Complete report on testing and development progress for repair weldments on irradiated materials, *on annual basis*
- Initiate fabrication of material for irradiated weldment testing, *June 2011– COMPLETED*
- Initiate irradiation of test plates with tailored helium concentrations for demonstration of weld technologies, *December 2012 – COMPLETED*.
- **Demonstrate initial solid-state welding on irradiated materials, December 2016**
- **Complete transfer of weld-repair technique to industry, August 2018**
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: Demonstration of advanced weldment techniques for irradiated materials (2016) is a key step in validating this mitigation strategy. Successful deployment (2018) may also allow for an alternative to core internal replacement and would be of high value to industry by avoiding costly replacements. Further, these technologies may also have utility in repair or component replacement applications in other locations within a power plant.

3.7.2 Advanced Replacement Alloys

Advanced replacement alloys for use in LWR applications may provide greater margins of safety and performance and provide support to industry partners in their programs through more economical operations. This task will explore and develop new alloys in collaboration with the EPRI Advanced Radiation-Resistant Materials Program. Specifically, the LWRs program will participate in expert panel groups to develop a comprehensive R&D plan for these advanced alloys. Future work will include alloy development, alloy optimization, fabrication of new alloys, and evaluation of their performance under LWR-relevant conditions (e.g., mechanical testing, corrosion testing, and irradiation performance among others) and, ultimately, validation of these new alloys. Based on past experience in alloy development, an optimized alloy (composition and processing details) that has been demonstrated in relevant service conditions can be delivered to industry by 2024. Current alloys of interest include austenitic 310 and 800 alloys, Ni-base 625 and 725, and ferritic martensitic grades T91 and 92. Additional testing is also being performed on 14YWT, an oxide dispersive strengthened 9-14 Cr ferritic steel. Phase I testing is currently concluding with high temperature steam oxidation testing, mechanical testing (including impact and fracture toughness testing), microstructural analysis, proton irradiation and preliminary SCC testing. Comparison of alloy performance to widely used alloy 316L in steam oxidation testing at 600 and 650°C is shown in Figure 21.

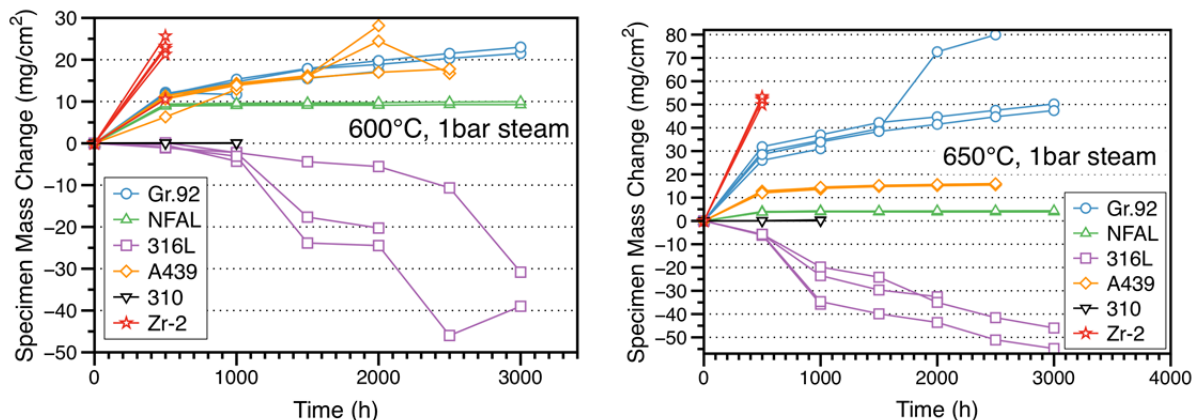


Figure 21: Comparison of the mass changes during high temperature steam oxidation testing of various alloys as compared to stainless steel 316L. The oxide dispersive strengthened 14YWT (NFAL) and high chrome A439 exhibit lower weight gains as compared to 316L and Zr-2.5Nb (Zr-2) [20].

Product: Development of new, advanced alloys for use in LWR applications and high quality data to support qualification

Lead Organization: ORNL, with support from University of Michigan

Current Partners: EPRI (cost sharing and partnership in Advanced Radiation Resistant Materials Effort) other partnerships included Bechtel Marine Propulsion Corporation and General Electric.

Project Milestones/Deliverables

- Complete report on testing progress, *on annual basis*
- **Complete down-selection and development plan in cooperation with EPRI, February 2013 - COMPLETED**
- Initiate collaborative research with EPRI on advanced alloys, *April 2013 - COMPLETED*
- Deliver characterization of select as-received advanced alloys as part of joint effort on Advanced Radiation Resistant Materials effort, *August 2014 - COMPLETED*
- Initiate ion-irradiation campaign to screen candidate advanced alloys, *January 2015 - COMPLETED*
- **Complete down-select of candidate advanced alloys following ion irradiation campaign, July 2017**
- Initiate neutron-irradiation campaign to test and validate advanced alloys, *October 2018*
- **Complete development and testing of new advanced alloy with superior degradation resistance with ARRM partners, September 2024**

Value of Key Milestones to Stakeholders: Completing the joint effort with EPRI on the alloy down-selection and development plan (2013) was an essential first step in this alloy development task. The development of advanced radiation-resistant materials may enable greater safety

margins and resistance to key forms of degradation at high fluences and long, component lifetimes.

3.7.3 Thermal Annealing

Post-irradiation annealing is still an approach of international interest to combat embrittlement migration, especially given the potential doubling or more of neutron exposure to be experienced with life extension to 80 years. Thermal annealing of RPVs has been demonstrated 15 times around the world, but not in the United States at full reactor scale. The NRC has issued a regulatory guide on thermal annealing of RPVs, but the nuclear industry has apparently been reluctant to adopt the procedure for non-technical reasons. Given operation of some very radiation-sensitive RPVs to 80 years, and considering the unknown factors discussed in this paper, it is likely that thermal annealing may be seriously considered in the future. Thus, there is a need for additional data on re-irradiation behavior of annealed RPV materials.

The thermal annealing task provides critical assessment of thermal annealing as a mitigation technology for RPV and core internal embrittlement and research to support deployment of thermal annealing technology. This task will build on other RPV tasks and extend the mechanistic understanding of irradiation effects on RPV steels to provide an alternative mitigation strategy. This task will provide experimental and theoretical support to resolving the technical issues required to implement this strategy. Specifically, this task will provide experimental testing and analysis related to the effects of reirradiation on annealed RPV materials (from the decommissioned Zion RPV and materials from the ATR-2 experiment). The same materials and test techniques used in other tasks will be applied here, extending the value of this work. Successful completion of this effort will provide the data and theoretical understanding to support implementation of this alternative mitigation technology.

Product: Development of annealing techniques, high quality data to support use of thermal annealing including annealing and reirradiation data, mechanistic understanding of reirradiation effects, and model capability for annealing (coupled with RPV task in Section 3.3.1)

Lead Organization: ORNL

Current Partners: NA

Project Milestones/Deliverables

- Complete report on testing progress, *on annual basis*
- Complete assessment of postirradiation annealing status and needs, and develop strategy plan for implementing postirradiation annealing, *September 2011 – COMPLETE*
- Initiate postirradiation annealing testing and evaluations on existing RPV specimen sets, *March 2018*
- Initiate reirradiation efforts on existing RPV specimen sets following annealing treatment, *April 2019*
- Perform demonstration of thermal annealing on RPV sections harvested from reactor, *April 2020*
- Complete reirradiation on RPV sections following thermal annealing, *September 2021*

- **Complete characterization of demonstration of RPV sections following annealing and reirradiation, *September 2025***
- Future milestones and specific tasks will be based on the results of the previous year's testing, as well as ongoing, industry-led research.

Value of Key Milestones to Stakeholders: While a long-term effort, demonstration of annealing techniques and subsequent irradiation for RPV sections is a key step in validating this mitigation strategy. Successful deployment may also allow for recovery from embrittlement in the RPV, which would be of high value to industry by avoiding costly replacements.

3.8 Integrated Industry Activities

Access to service materials from active or decommissioned nuclear reactors provides an invaluable access to materials for which there is limited operational data or experience to inform relicensing decisions and, in coordination with other materials tasks, an assessment of current degradation models to further develop the scientific basis for understanding and predicting long-term environmental degradation behavior. LWRS is currently engaged in two key activities that support multiple research tasks in the previous sections: the R.E. Ginna baffle bolt project and the Zion harvesting project.

The Zion Harvesting Project, in cooperation with Zion Solutions, is coordinating the selective procurement of materials, structures, components, and other items of interest to the LWRS program, ERPI, and NRC from the decommissioned Zion 1 and Zion 2 nuclear power plant, as well as possible access to perform limited, onsite testing of certain structures and components. Materials of high interest include low-voltage cabling, concrete core samples, and through-wall-thickness sections of RPV. For example, the acquisition of high value specimens from the RPV section (Figure 22) will go into supporting numerous tasks within LWRS and in other materials aging and degradation programs around the country. Currently, two large panel sections that contain the beltline weld have been sectioned from the decommissioned Zion plant and are now undergoing further sectioning through the end of 2016 and into 2017. Images of the panel locations relative to the RPV and from the sectioning operation work at Zion are shown in Figure 22.

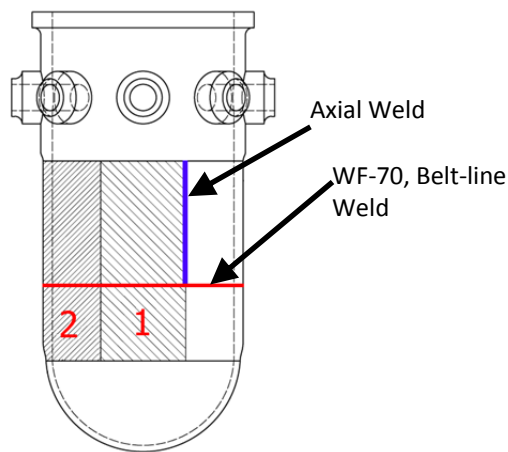


Figure 22: Diagram of the two panel sections harvested from the Zion Unit-1 RPV and photograph showing one of the panels being lowered into the railcar shipping box for transport to a secondary facility for further sectioning into test samples.

The other Integrated Industry Activity involves coordinating with R.E. Ginna Nuclear Power Plant (Exelon), Westinghouse Electric Company, LLC, and ATI Consulting, the selective procurement of baffle bolts that were withdrawn from service in 2011 and currently stored on site at Ginna. The goal of this program is to perform detailed microstructural and mechanical property characterization of high fluence baffle former bolts following in-service exposures. The bolts available are the original alloy 347 steel fasteners used in holding the baffle plates to the baffle former structures within the lower portion of the pressurized water reactor vessel. The two bolts selected for study were of the highest fluences available, but with overlapping fluence profiles across the length of the bolt. Damage values between the bolts range from 15 to 42 dpa, which correlate to levels in which limited data exists for many degradation phenomenon. Retrieval of the bolts is planned to occur in August of 2016, with inspection and sectioning of the bolts into test specimens through the first part of 2017. Testing will include the evaluation of fracture toughness and fatigue crack growth rates, microstructural analysis, in situ deformation studies under microstructural analysis and IASCC testing. The information from these bolts will be integral to the LWRS program initiatives in evaluating end of life microstructure and properties and are important for the benchmarking of models developed for radiation-induced swelling, segregation and precipitation. Furthermore, the material retrieved from Ginna can be used to compare against harvested material from other plants that have shown in-service IASCC damage.

4. Research and Development Partnerships

Effective and efficient coordination will require contributions from many institutions, including input from EPRI's parallel activities in the Long-Term Operations (LTO) program strategic action plan and NRC's Life Beyond 60 activities. In addition to contributions from EPRI and NRC, participation from utilities and reactor vendors will be required. Given the breadth of the research needs and directions, all technical expertise and research facilities must be employed to establish the technical basis in this R&D area for extended operations of the current reactor fleet.

The activities and results of other research efforts in the past and present must be considered on a continuous basis. Collaborations with other research efforts may provide a significant increase in cost sharing of research and may speed up research for both partners. This approach also reduces unnecessary overlap and duplicate work. Many possible avenues for collaboration exist, including the following:

- **EPRI:** Considerable research efforts on a broad spectrum of nuclear reactor materials issues that are currently under way provide a solid foundation of data, experiences, and knowledge. R&D cooperation on selected materials R&D activities is reflected in the LWRS Program/LTO Program Joint R&D Plan [21].
 - **Current collaborations:** IASCC, Environmentally Assisted Fatigue, Concrete Degradation, Cable Aging and Degradation, Advanced Replacement Materials (EPRI's Advanced Radiation Resistant Materials), Advanced Repair Welding, NDE technologies
 - **Additional potential collaborations:** Crack Initiation in Ni-base Alloys, swelling and high fluence phase transformations
- **NRC:** The broad research efforts of NRC should be considered carefully during task selection and implementation. In addition, cooperative efforts through the conduct of the Extended Proactive Materials Degradation Assessment and the formation of an Extended Service Materials Working Group will provide a valuable resource for additional and diverse input.
 - **Current collaborations:** EMDA, Concrete Degradation, and Cable Aging and Degradation
 - **Additional potential collaborations:** Crack Initiation in Ni-base Alloys, high fluence effects on RPV steels, material variability and attenuation effects, high fluence alloy performance
- **Boiling Water Reactor and Pressurized Water Reactor Owners Groups:** These groups provide a forum for understanding key materials degradation issues for each type of reactor.
 - **Current collaborations:** None
 - **Additional potential collaborations:** IASCC, Environmentally Assisted Fatigue, Concrete Degradation, Cable Aging and Degradation, Crack Initiation in Ni-base alloys, swelling and high fluence phase transformations
- **Materials Ageing Institute:** The Materials Ageing Institute (MAI) is dedicated to understanding and modeling materials degradation; a specific example might be the issue of environmentally assisted cracking. The collaborative interface with the MAI is coordinated through EPRI, which is a member of the MAI.
 - **Current collaborations:** None
 - **Additional potential collaborations:** Cable Aging and Degradation, Crack Initiation in Ni-base Alloys, swelling and high fluence phase transformations

- **Programs in other industries and sectors:** Research in other fields may be applicable in the LWRS program; for example, efforts in other fields such as the Advanced Cement-Based Materials program may provide a valuable starting point for developing a database on concrete performance for structures.
- **Nuclear facilities:** Examining materials from nuclear facilities provide a unique opportunity to evaluate degradation modes in relevant service materials. For example, the primary focus of the program centers on the material aging effects on RPV, core internals, concrete and cables. This is a significant program commitment. However, degradation of concrete, buried piping, and cabling is not unique to nuclear reactors; other nuclear facilities (such as hot cells and reprocessing facilities) may be a key resource for understanding long-term aging of these materials and systems.
- **Other nuclear materials programs:** In addition, research within fast reactor and fusion reactor programs may provide key insights into high-fluence effects on materials because the mechanisms and models of degradation for fast reactor applications can be modified and provide a starting and proven framework for degradation issues in this effort. This research element includes (1) international collaboration to conduct coordinated research with international institutions (such as Materials Ageing Institute) to provide more collaboration and cost sharing; (2) coordinated irradiation experiments to provide a single integrated effort for irradiation experiments; (3) advanced characterization tools to increase materials testing capability, improve quality, and develop new methods for materials testing; and (4) additional research tasks based on results and assessments of current research activities.

Participation and collaboration with all of these partners may yield new opportunities for collaboration. Cost sharing is being pursued for each task. Cost sharing can take many forms, including direct sharing of expenses, shared materials (or rescued specimens), coordinated plans, and complementary testing.

5. Research and Development Products and Deliverables

As described in Section 1, the Light Water Reactor Sustainability (LWRS) program is designed to support the long-term operations (LTO) of existing domestic nuclear power generation with targeted collaborative research programs into areas beyond current short-term optimization opportunities. Understanding the complex and varied materials aging and degradation in the different reactor systems and components will be an essential part of informing extended service decisions. The MAaD pathway is delivering that understanding of materials aging and degradation, providing the means to detect degradation, and overcoming degradation for key components and systems through new techniques.

As described in Section 1, the outcomes of the diverse research topics within the LWRS MAaD pathway can be organized into five broad categories:

- *Measurements of degradation:* High-resolution measurements of degradation in all components and materials are essential to assess the extent of degradation under extended service conditions, support development of mechanistic understanding, and validate predictive models. High quality data are of value to regulatory and industry interests in addition to academia.
- *Mechanisms of degradation:* Basic research to understand the underlying mechanisms of selected degradation modes can lead to better prediction and mitigation. For example, research on IASCC and PWSCC would be very beneficial for extended lifetimes and could build on other existing programs within EPRI and NRC. Other forms of degradation such as swelling and embrittlement are better understood, so mechanistic studies are not needed.
- *Modeling and simulation:* Improved modeling and simulation efforts have great potential in reducing the experimental burden for life extension studies. These methods can help interpolate and extrapolate data trends for extended life. Simulations predicting phase transformations, radiation embrittlement, and swelling over component lifetimes would be extremely beneficial to licensing and regulation in extended service.
- *Monitoring:* While understanding and predicting failures are extremely valuable tools for the management of reactor components, these tools must be supplements to active monitoring. Improved monitoring techniques will help characterize degradation of core components. For example, improved crack detection techniques will be invaluable. New nondestructive examination techniques may also permit new means of monitoring RPV embrittlement or swelling of core internals.
- *Mitigation strategies:* While some forms of degradation have been well researched, there are few options in mitigating their effects. Techniques such as post-irradiation annealing have been demonstrated to be very effective in reducing hardening of the entire RPV. Annealing may be effective in mitigating IASCC, based on initial studies. Water chemistry techniques such as NobelChem have been very effective in reducing some corrosion problems. Additional research in these areas may provide other alternatives to component replacement.

Every research task described in Section 3 delivers results in at least one of these categories. The outcomes and deliverables are detailed in Table 1 for each research task.

Table 1: Comparison of MAaD deliverables

Task Name	Measurements of Degradation	Mechanisms of Degradation	Modeling and Simulation	Monitoring	Mitigation Strategies
Project Management	NA	NA	NA	NA	NA
Assessment and Integration (EMDA)	NA	NA	NA	NA	NA
High Fluence Effects on RPV	□	□	□		
Material Variability and Attenuation	□	□	□		
IASCC	□	□	□		
High Fluence IASCC	□	□			
High Fluence Phase Transformations	□	□	□		
High Fluence Swelling	□	□	□		
Crack Initiation In Ni-base Alloys	□	□	□		
Environmental Fatigue	□	□	□		
Cast Stainless Steels	□	□			
Concrete	□	□	□	□	
NDE of Concrete				□	
Cable Degradation	□	□	□		
NDE of Cable Degradation				□	
Advanced Weld Repair	□		□		□
Advanced Replacement Alloys	□				□
Thermal Annealing	□	□	□		□
Baffle Bolts	□	□	□		
Zion	□	□		□	

As noted above, the strategic goals of the MAaD pathway are to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants and to provide data and methods to assess performance of SSCs essential to safe and sustained nuclear power plant operations. This information must be provided in a timely manner to support licensing decisions within the next 5–7 years. Near-term research is focused primarily on providing mechanistic understanding, predictive capabilities, and high-quality data. Longer-term research will focus on alternative technologies to overcome or mitigate degradation. The implementation schedule shown in Figure 23 is structured to support a number of high-level milestones. The value and impact of each of these milestones are described in detail in Section 3.

Industry	By 2017	By 2018 - First SLR License Application Submitted	By 2020 - First SLR license Approved by NRC	By 2024 - First License Renewal Expires
LWRS Program	Model for Cu-rich and Mn-Ni-Si precipitate development in RPV steels.	Model for transition temperature shifts in RPV steels.	Concrete performance model	RPV mitigation techniques and evaluation: annealing and post-anneal irradiation studies
	Combined thermal and radiation induced segregation / precipitation models	Mini-CT Specimen Development and Validation	Predictive degradation model for cables	Methods for Cable Rejuvenation
	Fatigue crack initiation and propagation modeling in reactor coolant system pipe base and weld material.	Transfer of weld repair techniques on irradiated materials to industry	Predictive model capability for IASCC susceptibility.	Development and testing of new advanced alloys with superior degradation resistance.

Figure 23: MAaD pathway implementation schedule and select key deliverables.

The key milestones of the MAaD pathway for 2016 and beyond are listed here.

2016: During 2016, mechanistic understanding of degradation in several materials systems, the establishment of the welding cubicle for irradiated materials and identification of NDE techniques that are suitable for monitoring cable degradation. Key milestones in 2016 will include:

- Complete key analysis of key degradation modes of cable insulation
- Complete assessment of cable mitigation strategies
- Complete assessment of cable insulation precursors to correlate with performance and NDE signals
- Demonstrate initial solid-state welding on irradiated materials
- Deliver unified parameter to assess irradiation-induced damage in concrete structures

2017: During 2017, there are numerous milestones for mechanistic understanding and model development for multiple material systems and components. Key milestones in 2017 will include:

- Provide model for transition temperature shifts in RPV steels
- Complete modeling of RPV steel hardening as a function of radiation flux, fluence, temperature and alloy composition
- Deliver computational tool to model combined thermal and radiation induced segregation of impurity solute elements to grain boundaries in austenitic stainless steels
- Initiate benchmarking testing for IASCC predictions using plant component materials
- Deliver predictive capability for swelling in LWR components
- Complete experimental validation and deliver model for environmentally assisted fatigue in LWR components
- Complete model tool to assess the impact of irradiation on structural performance for concrete components
- Complete down-select of candidate advanced alloys following ion irradiation campaign

2018: During 2018, there are numerous milestones for mechanistic understanding and model development for multiple material systems and components. Tech transfer of advanced repair welding techniques is expected in 2018. Other key milestones in 2018 will include:

- Establish validation of model for transition temperature shifts in RPV steels
- Deliver experimentally validated, physically based thermodynamic and kinetic model of precipitate phase stability and formation in Alloy 316 under anticipated extended lifetime operation of LWRs
- Complete analysis and simulations on aging of cast stainless steel components and deliver predictive capability for cast stainless steel components under extended service conditions
- Complete characterization of irradiation in concrete materials
- Complete prototype proof-of-concept system for NDE of concrete sections,
- Begin benchmarking of cable degradation model
- Complete characterization of repair welds on field component
- Complete transfer of weld-repair technique to industry

- Initiate neutron-irradiation campaign to test and validate advanced alloys

2019–2025: Longer-term R&D is expected to deliver key results in the 5–10 year window. Highlights will include delivery and deployment of NDE sensors, predictive modeling tools, and mitigation strategies.

- Complete analysis of copper variations through-RPV thickness and evaluate uncertainties with regard to irradiation-induced degradation and safety margins
- Deliver predictive model capability for IASCC susceptibility
- Deliver predictive model capability for Ni-base alloy SCC susceptibility
- Complete model tool to assess the combined effects of irradiation and alkali-silica reactions on structural performance for concrete components
- Deliver predictive model for cable degradation
- Deliver predictive capability for end of useful life for cable insulation
- Complete prototype of concrete NDE system
- Complete development and testing of new advanced alloy with superior degradation resistance with ARRM partners

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